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POWER REACTOR TECHNOLOGY

A Quarterly Technical Progress Review

Prepared for U. S. ATOMIC ENERGY COMMISSION by GENERAL NUCLEAR ENGINEERING CORP.

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● VOLUME 2

● NUMBER 3

TECHNICAL PROGRESS REVIEWS

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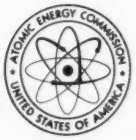
POWER REACTOR TECHNOLOGY

a review of recent developments *prepared by*

GENERAL NUCLEAR ENGINEERING CORP.

JUNE 1959 VOLUME 2 NUMBER 3



foreword

This quarterly review of reactor development has been prepared at the request of the Division of Information Services of the United States Atomic Energy Commission. Its purpose is to assist interested organizations in the task of keeping abreast of new results in reactor technology for civilian application.

The report is a concise discussion of selected phases of research and development for which there have been significant advances or a heightened interest in the past few months. It is not meant to be a comprehensive abstract of all material published during the quarter, nor is it meant to be a treatise on any part of the subject. The intention is to cover the various areas of reactor development from the general viewpoint of the reactor designer rather than from the more detailed points of view of specialists in the individual areas. However, papers which are thought to be of particular significance or particular usefulness in specialized fields will be mentioned in short notes. In the over-all plan of the report, it is intended that various subjects will be treated from time to time and will be brought up to date at that time.

Any interpretation of results which is given represents only the opinion of the editors' of the report, who are General Nuclear Engineering Corporation personnel. Readers are urged to consult the original references in order to obtain all of the background of the work reported and to obtain the interpretation of the results given by the original authors.

W. H. Zinn
General Nuclear Engineering Corporation

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GENERAL RESEARCH AND DEVELOPMENT

NUCLEAR FUEL RESOURCES AND FUEL UTILIZATION

Currently, the major problem facing the nuclear energy industry is that of producing nuclear plants that can compete economically with those using conventional fuel. Until this problem is solved nuclear energy will constitute only a small fraction of the total energy used. It is, nevertheless, worthwhile and necessary to survey periodically the magnitude of nuclear fuel reserves and the possible extent of the application of nuclear power in order to establish sound long-range goals for nuclear reactor development. In particular, judgment is affected by such considerations as to the degree of emphasis which should be placed on the early development of reactors and fuel cycles capable of high fuel utilization. Currently, judgments on this question differ,[1] as well they might, since they must be based on rather large extrapolations into the future. Of more concern to the reactor designer and builder is the question of whether reactor types currently under development make sufficiently effective use of nuclear fuel to assure their acceptability for some years or decades to come. The following discussion is aimed at presenting as many as possible of the pertinent facts and the assumptions which underlie such questions.

Structure of U. S. Energy Consumption

In 1956 the United States (plus Alaska) consumed a total of some 42×10^{15} Btu.[2] Petroleum supplied 44 per cent, coal 29 per cent, natural gas 23 per cent, and hydroelectric 4 per cent of this total. This energy was utilized approximately as shown in Table I-1.[3] It is difficult to see how nuclear power would be of interest in the fields of civilian propulsion (other than ship propulsion) and nonelectric residential heating. Accordingly, the areas of greatest opportunity for nuclear power seem to be the generation of electricity and industrial process and space heating. The rate of growth of electric power in the United States has been the subject of considerable comment. Table I-2 shows several estimated annual growth rates for the near future.[3] The growth rate of in-

Table I-1 ENERGY APPLICATIONS IN THE UNITED STATES,[3] 1956

Use	Per cent
Electric power generation	18.2
Industrial process heat	24.6
Propulsion	
Railroads	2.3
Ocean shipping	1.2
Military	2.6
Autos and airplanes	16.3
Residential and commercial heating	29.6
Waste, nonfuel use, etc.	5.2
Total	100.0

Table I-2 ESTIMATES OF RATE OF GROWTH OF ELECTRICAL CONSUMPTION IN THE UNITED STATES[3]

Estimator	Annual growth rate, %	Doubling time, years	Utility and industrial generation in 1980, kw-hr × 10^9
President's Material Policy Commission[4]	5.25	13.5	
Electrical World Magazine[5]	6.8	10.5	
McKinney Panel[6] (av.)	6	11.9	2210
Federal Power Commission[7]	5.0	14.2	1695
Philip Sporn[8]	6.7	10.5	
National Planning Association[3]	4.3	16.5	1795
	Av. 13 years		

dustrial heat consumed as process heat has been estimated in reference 3, for the time period 1954 through 1980, and the doubling time is about 20 years.

Fossil Fuel Resources

It is now a rather well-established fact that nuclear electric power, although somewhat more expensive than power which is produced under favorable circumstances from coal and oil, need not be more expensive by a large factor. In areas where the cost of fossil fuel is high, as in the United Kingdom, competitive nuclear power apparently may be foreseen within the present generation of reactors. Thus any depletion of the fossil fuel reserves which causes substantial increases in fuel prices may be expected to lead rather quickly to an important demand for nuclear power, even if such a demand has not already developed as a result of decreases in nuclear power costs. Hence the supply of fossil fuels is of interest to both the reactor designer and the conservationist.

It is generally agreed that the reserves of oil and natural gas represent a small store of energy compared to that of the coal reserves; consequently it suffices here to consider only coal reserves. A number of investigators (references 2 to 4, 6, and 9 to 11) have considered fossil reserves in relation to nuclear demand, and conclusions have in some cases appeared contradictory, partly because the estimates have not all been directed at the same question. Typical of the results which indicate a long life for the fossil fuel supply are those of reference 2:

"With the assumption that only half of the remaining reserves are actually recoverable as fuel, the continental United States reserves are found to have totaled approximately 892,025 million tons of coal as of Jan. 1, 1953. Since the production of coal during 1956—the last year for which accurate statistics are available—is reported as 527.8 million tons, it can easily be observed that the recoverable reserves are more than 1600 times as large as the 1956 annual consumption. Therefore, even though discrepancies exist in the reserve estimates, and annual consumption may increase, the volume of reserves is felt to be so large that they are adequate for several centuries."

A different approach is typified by the study of Putnam,[9] which is an exhaustive study made in 1953 under contract to the Atomic Energy Commission, to answer the question: "What is the maximum plausible role that nuclear fuels may be called on to play in the next 50 years or so?" Putnam emphasizes the point that in determining the demand for nuclear fuel the essential question is not what the reserves of fossil fuels are, but how much fossil fuel can be extracted at a given cost. In his words: "There is more coal, oil, and gas in the earth's crust than will ever be used. It is not a question of emptying the bin. It is only a question of deciding at what point it no longer pays to dig deeper." According to Putnam the trend in unit costs of extracting coal has already started up in the United States, and if by coal reserves we mean those which yield energy at a cost not grossly higher than the present, he would set the figure at about one-fourth that mentioned in reference 2. When he applies to this estimate the further estimate of the plausible increase in energy demand in the United States, which he arrives at by estimating the rate of growth of per capita energy demand (presently 3.4 per cent per year) and the rate of population growth, Putnam foresees the *possibility* that the reserves of coal recoverable at roughly present costs may be exhausted in the next 50 years.

Rate of Growth of Nuclear Electric Capacity

Evidently no substantial fraction of the electric power will be generated from nuclear fuel in the United States until the cost of such power is judged, by power producers, to be competitive with that generated from fossil fuel somewhere in the country. Thus any estimate of the rate of growth of nuclear capacity over the next few decades must be based on three subsidiary estimates, listed in order of decreasing importance: the rate at which development programs will reduce the cost of nuclear power, the rate at which fossil fuel costs will increase, and the rate of growth of demand for electric power.

Table I-3 lists several estimates of the nuclear generation of electricity in future years (from reference 3). The National Planning Association data on growth of nuclear genera-

tion of electricity were derived by Mayer and were also presented in a 1958 Geneva Paper.[15]

The factor of about 2 in per cent nuclear generation between the estimates of Davis and Roddis and of Mayer is especially interesting when the average costs of generating nuclear power used by the investigators are examined in Table I-4.

Table I-3 ESTIMATES OF NUCLEAR GENERATION OF ELECTRICITY IN THE UNITED STATES[3]

Estimator and date of estimate	Nuclear share of total annual generation, %		Total generation in 1980, 10^9 kw(e)-hr	Nuclear generation in 1980, 10^9 kw(e)-hr
	1980	2000		
Davis (1955)	35		2300	
Davis & Roddis[12] (1957)	37.8		3400	1285
Davis[13] (1959)		80		
McKinney Report[14] av. (1956)	19			
National Planning[3] Assoc. (1958); includes utility & industrial	21		1795	378

Table I-4 ESTIMATES OF NUCLEAR POWER COSTS

Investigator	Mills/kw-hr			
	1960	1965	1970	1975
Davis & Roddis[12]	12–20	9–14	8–10	7–8
Mayer[15]	15	9	6	5

Table I-5 ESTIMATES OF NUCLEAR ELECTRIC GENERATION IN THE UNITED STATES TO 1980, BY FIVE-YEAR PERIODS

Investigator	10^9 kw(e)-hr generated/year				Cumulative kw(e)-hr $\times 10^9$
	1965	1970	1975	1980	1965 to 1980
Mayer[3,15]	7.8	44.6	170.8	377.7	1900
Davis & Roddis[12]	22.8	134	526	1285	5700

These two projections will be used in the discussion that follows as approximate upper and lower limits of existing forecasts. The projections, by five-year periods, are given in Table I-5.

It may be seen that both sets of estimates assume that nuclear power becomes competitive within about the next 10 years, although there is perhaps a difference of opinion as to what power cost must (and can) be achieved in reaching this goal. No doubt every reactor designer and builder has his own opinions on these questions, and it may be that the estimates of Table I-5 would find wider acceptance if they were regarded as forecasts of nuclear power growth during the decade in which nuclear power becomes competitive and during the immediately following decade, rather than as forecasts for specific years. If so regarded the forecasts may lose much of their usefulness for predicting the rate of growth of the nuclear power industry, but they remain useful for estimating the fuel demands of the earliest families of reactors.

It may be remarked that a striking feature of the estimates is their very rapid rate of increase once competitive nuclear power is in sight. Such a characteristic is a result of the same circumstance that makes difficult the initial achievement of competitive plants: the cost of fossil fuel does not vary by a large factor over the United States. Thus the demand for nuclear power may be expected to be small until such power can be generated at something like the average cost of conventional power in the United States; thereafter the demand may increase very rapidly.

In Fig. 1 the estimates of total and nuclear annual electric generation, as estimated by Davis and Roddis and by Mayer, have been plotted as broad bands.

Also plotted in Fig. 1 are the outputs of power reactors built, building, or planned in the United States according to the December 1958 AEC Semiannual Report. These reactors are shown in Table I-6, and the conversion between capacity and output has been made by assuming a 90 per cent load factor. The agreement between the data in Table I-6 and the Davis-Roddis estimate is to be expected since the forecasters stated that they had included the plants in their estimate.

As an indication of nuclear plant capacity, the energy output estimates have been converted to plant capacity by the assumption of a 90 per cent load factor, and the result is represented in the figure by a band showing the equivalent number of 200 Mw(e) nuclear electrical plants. Since the assumed load factor is quite high, this represents the minimum plant capacity which would be consistent with the estimated electrical output. The electrical capacity of all nuclear

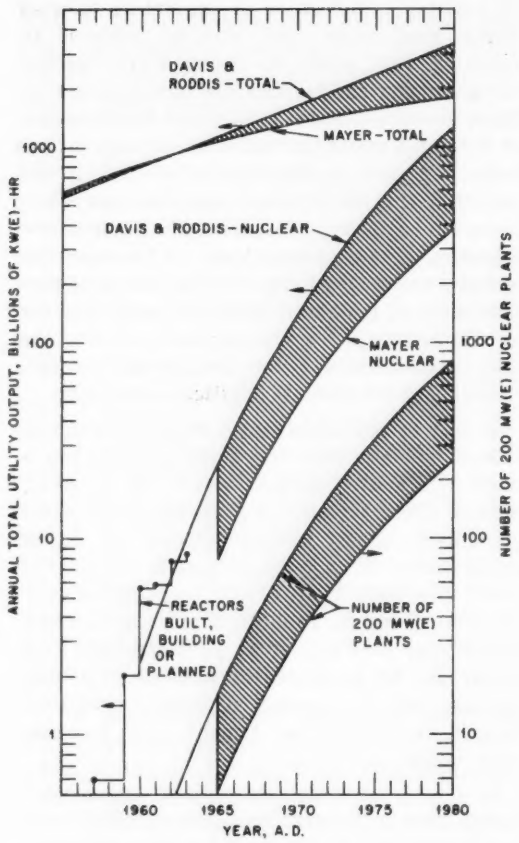

Figure 1 — Estimated growth of nuclear power generation.

electric plants presently built, building, or planned (Table I-6) is equivalent to about six 200 Mw(e) plants.

Estimates beyond 1980 are even more uncertain, for they depend upon extrapolations of total electrical demand which extend beyond two doubling times. Quite obviously the simple mathematical extrapolation of curves which are roughly exponential for more than about two doubling times can be subject to serious question. On the other hand, a forecast based upon considerations of economics, population growth, depletion of resources other than fuels, and technological developments is extremely complex, and for such considerations the reader is referred to the study of Putnam.[9] For the question raised in this Review, that of whether reactor types presently under development can anticipate fuel supplies sufficiently large to make them attractive ventures, the estimate to 1980 should suffice.

Table I-6 POWER REACTORS IN THE U.S.A.

Name	Year	Mw(e)	Kw(e)-hr/yr × 10^6	Cumulative kw(e)-hr/yr × 10^6
EBWR	1956	4.5	35.4	35.4
VBWR	1957	5.0	39.4	
Shippingport	1957	60.0	472	
SRE	1957	6.0	47.2	
	1957			594
Dresden	1959	180	1417	2011
EBR-II	1960	16.5	130	
Con. Edison	1960	255	2007	
Fermi	1960	90	708	
Yankee	1960	110	866	
	1960			5722
Elk River	1961	22	173	
Piqua	1961	10.5	83	
	1961			5978
Hallam	1962	75	590	
Northern States	1962	62.0	488	
Carolinas-Virginia	1962	17	134	
Chugach	1962	10.0	78.7	
Humboldt	1962	50.0	394	
Gas Cooled	1962	26.0	205	
	1962			7868
ECNG-FWCNG*	1963	58.0	456	8323

* East Coast Nuclear Group–Florida West Coast Nuclear Group.

Uranium Demands for Process and Space Heat

A recent, comprehensive report[2] covers the use of nuclear energy for the generation of process and space heat. The authors conclude that the potential for nuclear process and space heat is considerable. Data are also presented on the number of reactor installations that would occur if nuclear steam generation costs, which the authors believe possible, were achieved.

Generation costs were estimated for a boiling reactor of 20 Mw(t) capacity. Capital costs were estimated at about 1.7×10^6 dollars; the fixed cost, including operation and maintenance, was 54 cents per million Btu sent out. Fuel costs varied from 23 to 50 cents per million Btu depending on the worth of plutonium and enrichment of the fuel. The number of possible 20 Mw(t) reactor installations for process steam were computed to range from 323 to 792; the number of possible 40 Mw(t) reactor installations for space heating varied from 200 to 472.

These are stated to be minimum estimates and include semiquantitative factors to account for the preference of managements with regard to nuclear heat. The estimated total capacity of these plants thus ranges from about 14,000 to about 35,000 Mw(t), and if a plant factor of 50 per cent is assumed, the generation of steam would be something like 6×10^{10} to 15×10^{10} kw(t)-hr per year. Mayer[15] has estimated that potential U. S. sales of industrial process heat reactors may lie between 1000 and 7000 Mw(t) per year for the next 10 years. Teitelbaum[3] has projected the use of heat reactors assuming a steam cost of 35 cents per million Btu in 1965 for steam under 1000°F and 50 cents per million Btu in 1970 for steam at temperatures from 2500 to 3000°F. The generation of process heat by reactors during 1980 is estimated on this basis to be about 426×10^9 kw(t)-hr, with a capacity of about 53,000 Mw(t). The latter estimate, which appears to be the lowest of the three, would indicate a nuclear energy production for process heat in 1980 about one-third that of the lowest estimated production for electrical generation (Fig. 1). Inasmuch as no reactor has yet been used for process heat, there is little basis for choosing among estimates, but in view of the experience with nuclear electrical plants, most reactor builders would probably be inclined to accept the lower estimates, while investigating the potentialities of the process-heat field as a possible source of business.

Nuclear Fuel Reserves

It is a judicious time to consider how much uranium is present in the world at the present time. A more important question may be how much uranium is available to the United States. The world potential for the use of nuclear energy in the next few decades is greater than the potential in North America by a considerable amount, and the impact of this potential on the world's supply of uranium may be considerable. Data presented by Mayer[15] indicate that in 1970 Europe may be generating 50 to 60 per cent of the world's nuclear power in post-1960 plants in the price range from 10 to 15 mills/kw-hr. The nuclear generation of North America, at these price levels, is only a few per cent.

In a paper presented at the Second International Conference on the Peaceful Uses of Atomic Energy, J. C. Johnson, Director of the Division of Raw Materials of the U. S. Atomic Energy Commission, summarized the presently available information on nuclear fuel resources.[16] The estimated developed or partially developed reserves of high-grade uranium ore—defined as that from which uranium can be produced today at a cost of $10 per pound of U_3O_8 or less—amount to the equivalent of about 1.1×10^6 tons of U_3O_8 in the four countries for which reasonably good estimates are available: Canada, South Africa, United States, and France. The breakdown of the estimate by countries is given in Table I-7.

Table I-7 ESTIMATED RESERVES OF URANIUM ORE IN FOUR COUNTRIES

Country	Reserve, tons of U_3O_8	Date of estimate	Current production rate, tons of U_3O_8 per year
Canada	400,000	1957	13,000
South Africa	400,000	1957	6,000
United States	220,000	Current	15,000
France	50,000–100,000	1956	
Total	~1,100,000		

Allowing for conservatism in these figures,* and for probable reserves in Australia, the Belgian Congo, Portugal, and other countries, Johnson estimates that the total reserves in areas presently under development in the non-Communist world may be nearer 1.5×10^6 tons of U_3O_8 and that the ultimate production from these and adjacent areas may reach or exceed 2×10^6 tons. On the basis of present geologic data, and the discovery experience of the past 10 years, he predicts that an additional 2×10^6 tons of low-cost uranium should be developed in new areas within a reasonable period provided an aggressive exploration program is maintained. Of this, apparently at least 350,000 tons lie within the United States,† bringing the U. S. low-cost reserves up to 580,000 tons of U_3O_8.

The low-grade ore sources consist mainly of the bituminous shale and phosphate deposits, which typically contain 0.1 to 0.2 pound of ura-

*Reference 17, for example, indicates somewhat larger Canadian reserves.

†AEC staff paper quoted in reference 18.

nium per ton, in contrast with 2 to 6 pounds per ton for the high-grade ores. An estimate which considers only the large known deposits indicates a reserve of more than 20×10^6 tons of uranium in low-grade deposits over the world. Of this, five or six million tons is in the most uraniferous part of the Chattanooga shale in the United States, and more than 600,000 tons is in United States phosphate deposits. A limited research program for the U. S. Atomic Energy Commission indicated that uranium from the Chattanooga shale deposits would cost from $40 to $50 per pound if the shale were mined and processed for uranium alone.[16]

Thorium reserves in the non-Communist countries are estimated in reference 16 as possibly 500,000 tons, although the quantity could actually be considerably larger since there has as yet been no great incentive for thorium prospecting. Thorium will contribute importantly as an energy source only in reactors with high conversion or breeding ratios.

The uranium present in the United States can be utilized whenever desired. On the other hand, the deposits in Australia, Canada, France, the Belgian Congo, and other countries are subject to political agreements between nations and fall into somewhat the category of world petroleum reserves, where, for example, the United States has a direct interest in about 64 per cent of the free world's resources.[19] If the United States can count on important imports of uranium, then the quantity of low-cost uranium available may lie in the range 1 to 2 million tons or more; if not, the low-cost reserves apparently will amount to at least 600,000 tons, and the total reserves to at least six or seven million tons.

Degree of Fuel Utilization

Because of the unique circumstance that fertile isotopes (U^{238}, Th^{232}) may be converted to fissionable isotopes (Pu^{239}, U^{233}) in the nuclear reactor, the degree of fuel utilization may vary by a large factor from reactor to reactor. The degree of utilization is usually measured in terms of the specific energy production of the fuel—the number of megawatt days of thermal energy produced per ton of fuel passing through the reactor. If natural uranium is considered as the fuel, the destruction of all the contained U^{235} (some by fission, some by transmutation to U^{236}) in a thermal-neutron reactor will yield a thermal energy output of about 5000 Mwd/ton of natural uranium. The actual amount of energy obtained may be less or more than this quantity, depending upon the degree to which Pu^{239} is formed to augment the U^{235} and to help maintain reactivity.

For natural-uranium reactors, the specific energy production of the natural-uranium feed is just equal to the fuel-element lifetime measured in megawatt days per ton (if preirradiation processing and fabrication losses are neglected). It appears that only the D_2O-moderated reactor can achieve burn-ups longer than 5000 Mwd/ton with natural uranium, for only D_2O, of the common moderators, permits the use of natural uranium in the oxide form, which has high resistance to radiation damage. Fuel exposures as high as 8000 Mwd/ton or somewhat higher have been estimated for D_2O-moderated natural-uranium reactors employing oxide fuel and segmental reloading of the fuel.[20,21]

In the case of partially enriched fuel, the specific energy production, if referred to the natural-uranium feed of the isotope separation plant, may be very much less than the specific energy production of the fuel element.

By a simple material balance it may be shown that, if E is the enrichment of slightly enriched fuel, in grams U^{235} per gram of uranium, if X is the number of grams of U^{235} that has been extracted from each gram of uranium that is rejected from the plant as depleted uranium, and if N is the ratio of grams of uranium fed to the plant to grams of enriched product (enrichment E), then

$$N = \frac{E - 0.0072 + 1}{X}$$

where 0.0072 is the enrichment of the natural-uranium feed.

If, for example, E in the equation is taken as 0.015, and X as 0.005, then N is 2.56. Thus, if one desires to attain a specific energy production of 5000 Mwd/ton from the natural-uranium feed, by a single pass of the fuel element through the reactor, one must expose the 1.5 per cent enriched fuel element to a specific production of $5000 \times 2.56 = 12{,}800$ Mwd/ton.

The H_2O-moderated partially enriched reactors currently under development in the United States usually have, as target exposures for their fuel elements, a specific energy pro-

duction of 10,000 Mwd/ton, and there are many who believe that considerably longer exposures can be attained. It therefore seems reasonable to conclude that specific productions of about 5000 Mwd/ton of natural-uranium feed may characterize the water reactors currently under development. For reactors which use fuel of enrichment higher than about 1.5 per cent, this utilization will be achieved only by re-enriching and recycling the fuel through the reactor. This is, of course, the type of operation that is contemplated. For such a case, a utilization of 5000 Mwd/ton implies that a sufficient quantity of converted plutonium is burned to compensate for the U^{235} which is discarded with the low-enrichment stream from the diffusion plant. In such a fuel cycle, long fuel-element lifetime (at a given enrichment) favors high utilization because it implies a relatively high ratio of plutonium fissions to U^{235} fissions.

Very long fuel exposures are thought to be possible in reactors which dilute the fuel with moderator and which have a sufficiently high conversion ratio to maintain the reactivity over the necessary long exposure time. The proposed British high-temperature gas-cooled reactor[22] is an illustration of this concept.

Although it is only in reactors of special types that large increases in fuel utilization can be achieved through increased fuel-element life, it is to be expected that some improvement will be attained in most reactor types, for the fuel lifetime has a strong effect on fuel cost, and consequently there is a strong economic incentive to lengthen the lifetime.

The most probable method by which substantial increases in utilization will be attained is through the recycle of bred fissionable material (plutonium or U^{233}). Recycling will make substantial increases in the fuel utilization for any reactor which has a reasonably high conversion ratio. For reactors having conversion ratios of approximately 0.8., the recycling of plutonium can, on theoretical grounds, be expected to increase the specific energy production of the basic feed material by something like a factor of 4, and with still higher conversion ratios the theoretical specific production increases rapidly.[23] There are, however, practical limitations, such as the metallurgical life of fuel elements, which may limit the gains in individual cases. Although conversion ratios approaching 0.80 are not achieved in the pressurized-water reactors currently under construction in the United States, such a ratio can probably be attained when zirconium is used as fuel jacket and structural material if a relatively high ratio of fuel to water moderator is used in the reactor core. Substantially higher conversion ratios should be obtainable with the less absorbing moderators, such as graphite and D_2O, provided the absorption by structural materials is kept low. If U^{233} is used as fuel with the low absorption moderators, the conversion ratio can approach 1 or possibly become slightly greater than 1. High conversion ratios may also be attainable in H_2O-cooled and -moderated reactors having a very high ratio of uranium fuel to water in the core, and in some other "under-moderated" reactor types which use thorium as the fertile material. However, these types may require rather high enrichments and hence represent no gain in the utilization of the basic natural-uranium feed.

Reference 24 deals with the concept of using plutonium recycle in Calder Hall type reactors. The major results, briefly, are as follows:

1. Recycle increases the burn-up, based on reactivity, from a calculated value of 2749 Mwd/ton for no recycle, batch loading, to a value of 8763 Mwd/ton with five recycles, also assuming batch loading.

2. The most advantageous fuel-cycle technique, based on fuel costs, appeared to be the one wherein recycled plutonium is blended with natural uranium for the next run (no seed or spike techniques). Pertinent information on this recycle method is summarized in Table I-8.

Table I-8 FUEL-CYCLE PARAMETERS

Run No.	Duration, years	Life, Mwd/t	Fabrication costs of Pu-U elements, $/kg	Fuel-cycle costs, mills/kw(e)-hr
1	4.7	2749	10 (no Pu)	3.27
2	9.4	5428	20	2.87
			35	3.53
3	13	7517	20	2.46
			35	2.64
4	14.5	8391	20	2.36
			35	2.86
5	15.1	8713	20	2.33
			35	2.82
6	15.2	8763	20	2.33
			35	2.83

It should be pointed out that the shortcomings of uranium-metal fuel elements at high temperatures would prevent the practical attainment of the long fuel lifetimes of Table I-8. The

figures do indicate, however, how recycle may extend fuel utilization when practical considerations permit.

Reference 23 has considered the economics of recycle, and the authors are optimistic about the role plutonium recycle will play in large-scale nuclear power systems. Detailed fuel-cycle analyses have been performed in reference 25 and fuel costs for recycle and nonrecycle operation were found to be not appreciably different; for plutonium valued at $12 per gram, minimum fuel cost was associated with heavy-water reactors fueled with natural or slightly enriched uranium. These authors studied plutonium recycle in heavy-water reactors using natural-uranium feed. The burn-up (based on reactivity) obtainable as a function of cycle number is given in Table I-9.

Table I-9 BURN-UP CONSIDERATIONS IN D_2O-NATURAL-URANIUM REACTORS

Cycle No.	Maximum burn-up (based on reactivity), Mwd/t
1	4,850
2	9,600
3	12,400
4	13,180
5	13,200

These figures were calculated for the case of spatially uniform flux and batch charge-discharge. Although spatially varying flux reduces maximum burn-up, this effect can be more than compensated by going to nonbatch charge-discharge programs such that the average burn-up is spatially independent. This is the "uniformly graded irradiation" technique of Lewis.[21]

Detailed studies of plutonium recycle in light-water-moderated reactors[25,26] have been aimed at discovering methods of attaining low fuel cost under conditions which will prevail over the near future, rather than specifically at finding methods of extending fuel utilization. A rather large number of recycle variables were investigated in reference 26 as applied to a boiling reactor having the characteristics shown in Table I-10.

A regional charge-discharge technique was assumed whereby one-fifth of the reactor core was replaced at a time and the fuel was rearranged so as to approach a uniform mixture.

Table I-10 REACTOR PARAMETERS USED FOR RECYCLE CALCULATIONS

Reactor type: Boiling water
Pressure: 1015 psia (546.4°F saturation)
Core size: Cylinder 10.3 ft diameter by 8.6 ft length
Power: 630 Mw(t)

Compositions by volume: For volume ratio of 2.0 water/fuel
Fuel volume fraction: 0.30728
Water volume fraction: 0.61455
Cladding volume fraction: 0.07817
Average steam void fraction: 15% of water volume

Rod size: 0.494 in. (cold)
Cladding: 0.030 in. Zircaloy-2
Fuel: UO_2 with up to 1.4% of PuO_2 at 95% of theoretical UO_2 density
Chemical and fabrication losses: 0.4% per cycle

The longest fuel exposure investigated was 10,000 Mwd/ton. Thus, unless both plutonium and uranium were recycled, the basic fuel utilization was limited to 10,000 Mwd/ton even if such an exposure could be attained using natural-uranium feed. Actually, the exposure attainable with natural-uranium feed appeared to be limited to about 4000 Mwd/ton at best.

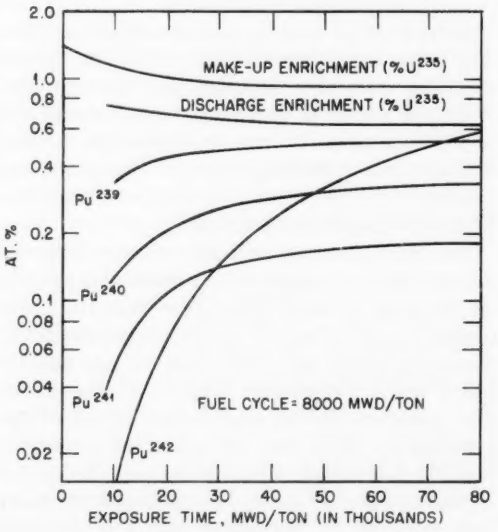

Figure 2—Initial enrichment and discharge fuel compositions.

The limitations imposed on fuel utilization by the finite metallurgical lifetime of the fuel are illustrated in Fig. 2, reproduced from reference 26. The situation to which the figure applies is

that in which the reactor is started initially on a loading of uranium only, enriched to the degree required to give a fuel life of 8000 Mwd/ton, a value arbitrarily chosen and held constant over the entire recycle operation. The initial enrichment, from the curve, is about 1.5 per cent. After the first cycle of 8000 Mwd/ton, one-fifth of the fuel is removed and the plutonium is extracted and added to a new batch of partially enriched fuel of lower enrichment than the initial loading, as required for a life of 8000 Mwd/ton. The fuel is then rearranged to give a uniform mixture. After several repetitions of this process, the enrichments of the new fuel loads (make-up enrichment) and of the discharged fuel (discharge enrichment) become nearly constant, at about 0.95 and 0.65 per cent, respectively. The figure shows also the build-up of plutonium isotopes in the fuel.

In the pseudoequilibrium condition the burn-up of U^{235} is 0.0095 − 0.0065 = 0.0030 g per gram of fuel. This corresponds to a specific energy production of about 2100 Mwd/ton. Thus the recycle of plutonium has multiplied by almost a factor of 4 the energy output of the U^{235}. It has not, however, resulted in a higher basic fuel utilization if the fuel discharged from the reactor must be discarded. Since the enrichment of this fuel is higher than the low-enrichment discharge of the diffusion plant, it should obviously be re-enriched and re-used. The true basic utilization can be most easily computed by assuming that the same U^{238} is recycled continuously through the reactor and that enough fully enriched U^{235} is added at the end of each cycle to bring its enrichment back to the initial value. Thus 0.003 g of U^{235} must be added to each gram of fuel per cycle. If 0.005 g of U^{235} can be extracted by the diffusion plant from a gram of natural uranium, then the new natural uranium used per cycle amounts to 0.003/0.005 g per gram of reactor fuel, or $\frac{3}{5}$ ton per ton of fuel, and the basic utilization is 8000 divided by $\frac{3}{5}$, or 13,300 Mwd/ton.

Very large gains in fuel utilization can, of course, be made by utilizing breeder reactors. Breeder reactors which have sufficiently high breeding gains to compensate for losses in chemical processing make possible the ultimate fissioning of a large fraction of the atoms in the fuel feed material, the only losses being those to the formation of higher isotopes of plutonium or uranium which are neither fissionable materials nor good fertile materials (e.g., Pu^{242}), and to chemical processing losses. Breeding has been shown to be technically feasible in fast-neutron reactors. The problems which remain for this reactor type are economic ones and those connected with plutonium recycle. The aqueous homogeneous reactor operating on the U^{233}-thorium cycle with D_2O moderator has the fundamental capability of breeding, but the realization of this potentiality is as yet involved with the technical problems of the reactor type.

Another rather recent entry into the thermal breeder field is the gas-cooled reactor. Perry[27] has considered both graphite- and D_2O-moderated reactors and concluded that the graphite reactor (which used graphite as a structural material also) shows little promise to do more than break even. The D_2O reactor, with zirconium pressure tubes containing graphite impregnated with U^{233}, showed more promise as a breeder but it was not apparent whether short doubling times (i.e., 10 years) could be achieved. A further question in connection with thermal breeding is the present uncertainty in the value of η of U^{233}, which was discussed in the previous issue of this Review.[28]

No studies of the precise degree of fuel utilization which might practically be expected with breeder reactors are known. Putnam assumes the feasibility of "burning" one-third of the metal mined. This would correspond to about 250,000 Mwd/ton of uranium or thorium. There is no apparent technical reason why a utilization as high as this or higher could not be achieved.

The subject of fuel utilization can perhaps be summed up as follows. The reactors presently under development may be expected to extract about 5000 thermal megawatt days from a ton of natural uranium. This figure could be perhaps twice as high for reactors of high neutron economy but hardly much more than 10,000 Mwd/ton without plutonium recycle. With recycle, the utilization may range perhaps from 10,000 Mwd/ton for the reactors of poorer neutron economy to quite high figures for, say, D_2O-moderated reactors, but many of the reactors now in the prototype stage will probably operate in the range from 10,000 to 20,000 Mwd/ton. With breeder reactors, which of course must also recycle bred material, the utilization could amount to 250,000 Mwd/ton or more.

Consumption of Uranium Resources

By combining the estimates of nuclear power production with those of fuel utilization, a forecast of nuclear fuel depletion may be made. In Table I-11 the estimates of Fig. 1 have been used to predict a possible situation in 1980. The table shows, for both the "high" and "low" estimates of Fig. 1, the installed nuclear electrical capacity (computed on a 90 per cent load factor), the nuclear electrical generation in 1980, the integrated generation through 1980, and the total generation commitment, including both the integrated generation through 1980 and the future generation by all plants existing in 1980, assuming a 20-year plant life. Energy demands for uses other than electrical generation have not been included, for it is felt that estimates for other commercial applications must be quite uncertain until some reactors have been built for those applications. As indicated earlier, it is conceivable that these demands could be substantial, possibly amounting to as much as one-third of the demand for electrical generation.

Table I-11 NUCLEAR FUEL COMMITMENTS IN 1980

	High	Low
Installed nuclear power capacity, millions of kw(e)	227	57.6
Billions of kw(e)-hr generated in 1980	1286	378
Billions of kw(e)-hr generated, 1965 to 1980	5980	1900
Billions of kw(e)-hr committed by a 20-year plant life	16,950	5070
Tons U_3O_8 "consumed" over life of plants built to 1980 at 30% efficiency:		
At 5000 Mwd/t	75×10^4	23×10^4
At 20,000 Mwd/t	18.8×10^4	5.8×10^4
At 250,000 Mwd/t	15,000	4600
Tons U_3O_8 in inventory in 1980 at 1 Mw(e)/t	22.7×10^4	5.8×10^4

The U_3O_8 consumptions implied by the above figures are given for three specific energy productions: 5,000, 20,000, and 250,000 Mwd/ton. The U_3O_8 in inventory is also shown, estimated on the basis of 1 Mw of installed electrical capacity per ton of U_3O_8.*

It is evident that, if the "high" estimates are used, a fuel utilization as low as 5000 Mwd/ton will result in the use of an amount of uranium comparable to our low-cost domestic reserves. It is hardly to be imagined, however, that such a low utilization would characterize such an extensive generation program.

Even at a specific energy production of 20,000 Mwd/ton, the higher estimate would show a large fraction of the presently known U. S. low-cost reserves committed by 1980. It must also be recognized that, once the expansion of nuclear power has reached a rate equal to that predicted for 1980 by the estimates of Fig. 1, the total fuel commitments will increase rapidly, and within the following decade or two high fuel utilization will become a necessity.

It is pertinent to ask just what effect the "consumption" of an amount of uranium equal to a large fraction of the U. S. low-cost reserve would have on the long-term fuel resource of the nation. If imports are large, the effect will of course be proportionately reduced. It is to be remembered that the use of uranium at a low specific energy output does not preclude the possibility of re-using the depleted material in a breeder reactor. The depletion of the uranium amounts to the loss of much conveniently stored U^{235}. It does not represent a large depletion of the potential fuel for breeder reactors, although it may conceivably limit the rate at which breeders can be supplied with an initial inventory of fissionable isotope for starting the breeding cycle. It may also deplete the supply of U^{235} available for reactor applications in which breeding is impractical. Reactors for ship propulsion and for process heat may be in this class.

A second consideration is the economic feasibility of using the high cost reserves. At a fuel utilization of 5000 Mwd/ton, comparable to the present utilization, and at the present uranium recovery cost of $10 per pound, this recovery cost makes a contribution of about 0.6 mill/kw-hr to the fuel cost. This contribution is not large. If the specific energy production were increased to 20,000 Mwd/ton, then a uranium recovery cost of four times the present

* For a different approach to the consumption and inventory problems, reference 18 should be consulted.

value would result in the same contribution to fuel cost. Thus it is apparent that as fuel utilization is increased the industry can afford to pay higher and higher prices for raw uranium, and the necessity of using the high cost uranium sources would probably not constitute a national economic tragedy. However, if uranium prices increase, an economic pressure will arise which will stimulate the development of reactor systems giving high fuel utilization.

In considering the figures of Table I-11, the reactor manufacturer may be struck by two important considerations. The first is that which has been considered at some length—that fuel utilizations higher than those which characterize for present thermal reactor designs may become necessary soon after the installed nuclear capacity reaches the magnitude indicated by the estimates of Fig. 1 for 1980. The second, and more striking fact, is that the estimates represent a very high level of installed capacity—the equivalent of between 250 and 1000 plants of 200 Mw(e) capacity by 1980. This represents a tremendous investment in nuclear capacity. At a specific cost of $200 per installed kilowatt, the investment in power plants alone would amount to some 10 to 40 billion dollars, and in addition there would necessarily be a large investment in processing plants, fuel fabrication plants, and other back-up facilities. Furthermore, the number of installed units is sufficiently high that several reactor types could be brought to economic maturity and yield return on the development investment. It seems evident that the conclusion of the manufacturer, if he believes in the possibility of such an explosive growth of the nuclear power industry, will be that his first concern must be to assure himself a share of the large market by developing as quickly as possible that reactor which can produce the cheapest power under the present conditions of fuel supply. If he is prudent, he will also recognize the need for developments leading to high fuel utilization in the near future, but he is apt to conclude that, if he does enjoy an important fraction of the large market, he will be in a position to carry out those developments successfully.

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APPLICATIONS: NUCLEAR PROPULSION FOR MERCHANT SHIPS

A number of recent studies of nuclear propulsion for merchant ships were discussed in the last issue of this Review.[1] More recently the proceedings of the 1958 Nuclear Merchant Ship Symposium[2] (held at Washington, D. C., on Aug. 21, 1958) have been published. The papers presented at the symposium cover three main subjects: the design of the N.S. *Savannah* and its power plant, the American-Standard study[3] of merchant ship propulsion, and work related to that study, and descriptions of proposed nuclear ship propulsion plants and installations. The first two of these subjects were covered in the previous Review, although the symposium proceedings cover certain additional information relative to the American-Standard study which is of interest to the designer of propulsion plants for commercial vessels. For example, papers three[4] and five[5] of the symposium, which cover economic analysis and ship design for conventional ships, have interesting background information relative to the nonnuclear factors which effect shipping economy. The following discussion will be confined primarily to those papers of the symposium which covered proposed designs of nuclear propulsion machinery.

Two of the papers described a study of the boiling-water reactor for ship propulsion, carried out by General Electric[6] and George G. Sharp, Inc.[7] The portion of the study which compared the performances of several types of boiling reactors was made in terms of a type T5-S-RM2a tanker. This vessel, which is not considered to be necessarily the optimum vessel for nuclear propulsion, is a single screw tanker of 31,400 tons displacement, utilizing 20,000 SHP normal, 22,000 SHP maximum, to attain a design speed of 20 knots. It is pointed out that this speed is higher than normal for a tanker of this size.

Three types of boiling-water nuclear power plants were considered: forced circulation direct cycle, natural circulation direct cycle, and forced circulation indirect cycle. Of these the natural-circulation direct-cycle plant was found to be most attractive. Table II-1 is the comparison of estimated weights and costs of the three boiling reactor types, as well as those for a conventional propulsion plant, and for a pressurized-water plant of the *Savannah* type. The implication of the reference is that the pressurized-water design is essentially that of the *Savannah* itself. It is therefore not clear whether the pressurized-water boiling-water comparison should be considered a comparison between reactor types or a comparison of the results of a construction design for the first nuclear propulsion reactor with those of a design study for a later reactor. As a further comparison of boiling and nonboiling plants, the comparable figures from the American-Standard study[3] have also been included in Table II-1. Although the two studies cover ships of rather different sizes, the power requirements are practically equal. There is, however, a marked difference between the two studies in the ratio

$$\frac{\text{Cargo per ton cost for pressurized-water ship}}{\text{Cargo per ton cost for boiling-water ship}}$$

the ratio being 1.73 in the George G. Sharp[7] study and 1.02 in the American-Standard[3] study. The discrepancy would indicate that the two studies used different approaches to the pressurized-water estimate.

In reference 6, the design of the natural-circulation direct-cycle boiling reactor which was selected as the most desirable in the comparative propulsion study[7] is described. It has a thermal rating of 59.7 Mw, operates at 1000 psig, and gives a net plant thermal efficiency of 27.6 per cent. It uses 2.1 per cent enriched uranium to attain a predicted fuel lifetime of 10,000 Mwd/ton.

Reference 6 also describes an economic study of this reactor when applied to a tanker of larger size (39,000 DWT, 50,000 tons displacement) than that of the previous study, an application which is believed to be more typical of tanker practice. In this case the estimated specific cost of the nuclear ship is considerably lower amounting only to $412 per cargo ton; the estimated ship cost ($15,400,000) is, in fact, lower than those

Table II-1 COMPARISON OF WEIGHTS, CAPACITIES, AND COSTS FOR VARIOUS NUCLEAR TANKERS

	George G. Sharp Study[7]					American-Standard Study[3]		
	1 Conventional steam	2 PWR Savannah type	3 BWR forced circulation	4 BWR forced circulation	5 BWR natural circulation	6 Conventional steam	7 PWR	8 BWR natural circulation
Cycle	Direct	Indirect	Direct	Indirect	Direct		Indirect	Direct
Shipweight, long tons:								
Steel	6430	6555	6520	6540	6620	7481	7481	7480
Outfit	1360	1320	1290	1320	1290	1205	1205	1205
Machinery	970	955	720	780	720	911	748	748
Reactor		480	170	250	170		879	885
Containment		200	170	310	170		†	†
Shielding								
Primary		110	850	850	850		†	†
Secondary		1830	*	1000	*		†	†
Total	8760	11,400	9820	11,050	9820	9597	10,303	10,318
Deadweight, long tons:								
Cargo	19,580	19,080	20,660	19,430	20,660	34,392	33,530	33,530
Fuel oil	2460	320	320	320	320	Incl. in cargo	140	140
Crew, etc.	600	600	600	600	600	347	347	347
Total	22,640	20,000	21,580	20,350	21,580	34,739	34,017	34,017
Costs in thousands of dollars:								
Ship	12,700	26,700	17,800	18,900	16,750	14,208	19,910	19,450
Cost/ton of cargo	649	1400	862	972	810		594	580
Length, over-all, ft	615	These values are based on the conventional ship data, column 1.				707	These values are based on the conventional ship data, column 6.	
Nominal SHP	20,000					20,200		
Sustained speed, kt	20					18		

*Included in deadweight.
†Included in reactor and/or machinery weight.

for the smaller vessels of reference 7 (Table I). The following conclusions were reached from the economic analysis:

1. Based on the assumptions employed, a boiling-water reactor propelled tanker initially operating in 1961 offers economic performance within 5 per cent of a conventionally propelled tanker.

2. Considering the long-range trend in fuel oil costs, this first tanker may be economically competitive with conventional propulsion in 5 to 10 years.

3. The capital charges and interest charges for nuclear propelled ships may always be higher than the corresponding costs for conventionally propelled ships. To offset these higher operating costs, a nuclear ship design must take advantage of the lower fuel costs and increased cargo capacity.

An essential ingredient of the conclusions, as mentioned, is the estimated lower fuel cost for the nuclear plant. The fuel cost estimates of the economic study were based on an estimated fuel fabrication cost of $70 per pound of uranium and a plutonium buy-back price of $30 per gram for the immediate future. The total fuel cost on the basis of these figures was estimated at 2.9 mills/SHP-hr (equivalent to $25.70 per Mwd(t)), as compared to 4.4 mills/SHP-hr for a conventional ship using fuel oil at $3 per barrel. As the plutonium buy-back price is reduced in future years to about $12 per gram, the fuel fabrication cost is predicted to decrease to about $40 to $45 per pound of uranium, to yield a nuclear fuel cost of 3.19 mills/SHP-hr ($28.30 per Mwd(t)).

The simplicity of the natural-circulation direct-cycle boiling reactor and the relative ease with which it may be shielded are factors which contribute heavily to its attractiveness for ship propulsion. Since this reactor type depends on gravitational forces and coolant density gradients for circulation of the coolant, it is natural to inquire whether the roll and pitch of a ship might not interfere seriously with its performance and stability. This question is not treated in either of the two papers of the symposium,

but the question was asked as part of the discussion. The question, and the answer supplied by V. A. Mize, of the General Electric Company, are quoted below:

"Given the effect of ship's pitch and roll on the velocity of rising steam bubbles, how does a marine boiling-water reactor plant maintain constant reactivity, i.e., constant output, when seas are running?

"*Answer:* Based on the design conditions of roll and pitch established for the T-5 tanker, a variation in steam bubble formation and hence reactivity of 0.13 per cent will occur in a sinusoidal manner. This change in reactivity is roughly equivalent to moving one of the 21 control rods 2 in. The effect can be considered negligible.

"As for the 'sloshing' of the water above the chimneys, the effect of the oscillation of the liquid level in the core would be to decrease the core void fraction by a factor of less than 1000/1002. This effect can be considered negligible."

Evidently the relatively high operating pressure of 1000 psi has an important bearing on the expected insensitivity of the reactor to inertial affects. Questions related to this have been treated in reference 8, which has been previously discussed in this Review.[9] However, a report of a detailed analysis of this question, aimed directly at the performance of boiling reactors in mobile installations, would be of considerable interest to the reactor designer.

An analysis of the transient performance of forced-circulation boiling reactors in ships is reported in reference 10. The authors have considered the effects of vertical acceleration on moderator specific gravity from the standpoint of bubble size, the length of the nonboiling region, change of slip velocity, and several other parameters. The parameter most sensitive to changing vertical acceleration was found to be the slip velocity. The mean core coolant specific gravity was reported to change by 0.43 per cent for a ship running in "considerable swells." If reactor power were reduced 40 per cent in this sea state, the effect of the acceleration was found to increase about 30 per cent. For a vertical acceleration of ±0.25 g with a 9-sec pitch period, the neutron number density was found to undergo a 3.8 per cent perturbation. The effect of the list of the ship was also studied and found to be relatively insignificant compared to the vertical acceleration effect.

Although the analysis was made for the forced-circulation case, the authors did consider the natural-circulation reactor. They concluded that ". . . it is advisable to adopt forced circulation on a ship despite the expense . . . ." This conclusion may well depend upon the specific design considered—in this case the natural-circulation reactor had no chimneys or stacks above the core, and a low degree of subcooling of the inlet feed water, with an attendant high recirculation ratio, was recommended. However, the concern expressed for the effects of ship motion on natural circulation does emphasize the need for a comprehensive treatment of this question.

An earlier, but recently published, study[11] of the application of a 75-Mw forced-circulation boiling reactor to a 38,000 DWT, 50,000-ton displacement, 22,000 SHP tanker reached conclusions which were less optimistic. This study, by American Machine and Foundry Company (AMF), estimates a specific ship cost of $558 per cargo ton ($20,765,248 for the ship, of which about $3 million is for escalation, estimated changes, owner-furnished equipment, and inspection and legal costs—items which are not mentioned in references 6 and 7). The fuel cycle considered in reference 11 is interesting in that regional charge-discharge is used. The original core is divided into five units and one-fifth of the core is replaced at a time. Seven fuel cycles were studied lasting over a time period of 9.33 years. The total cost over the seven cycles was reported as 5.801×10^6 dollars for a plutonium credit of $12 per gram and a U^{233} credit of $15 per gram. Dividing by the number of thermal megawatt days produced yields an average cost of $26.80 per Mwd(t). The segmental charge-discharge method, however, does introduce bookkeeping difficulties in the nonequilibrium period, since it is possible to either amortize the initial core costs or treat them as inventory against which interest charges are paid. The fuel cost for equilibrium operation was stated to be $32.50 per Mwd(t).

On the basis of these estimates, reference 11 suggests that: "In order to reach the point in nuclear technology and equipment where nuclear ships can compete economically, it may be necessary to subsidize the construction of nuclear ships as long as they remain at an economic disadvantage." An advantage of the General Electric reactor compared to the AMF reactor, at least with respect to fuel costs, is

the high plant efficiency (27.6 per cent compared to 22 per cent).

For a 75-Mw boiling-water reactor, reference 3 shows a fuel inventory charge of $525 per day plus variable fuel cost of 0.99 mill/kw(t)-hr based on 1957 prices. The variable fuel cost consists of fabrication, reprocessing, burn-up, loading, unloading, and shipping. For a load factor of 0.85 this amounts to $28.40 per Mwd(t). A plutonium buy-back price of $30 per gram is assumed, but since the steel-clad elements do not give the reactor a high conversion ratio, the fuel cost probably would not be effected much by a decrease to $12 per gram.

The gas-cooled marine propulsion reactor program was covered in a paper published as reference 12; more recent information has been obtained, however, and can be found in Sec. IX, this Review.

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*Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, September 1958. (Individual papers available from the Office of Technical Services, Dept. of Commerce, Washington 25, D. C. Bound volumes of papers will be available from United Nations, Office of Public Information, New York.)

Physics of Heavy-water Lattices

A number of recent papers[1–5] summarize experimental results obtained in investigations of the physics characteristics of natural-uranium lattices in heavy water. A wide variety of metal- and oxide-fueled lattices are covered in these experiments, including a wide range of fuel cluster geometries which are of practical interest in the attainment of high reactor specific power outputs. The principal objective of the experiments was the measurement of material bucklings, B_m^2. Some measurements were also made of the intracell distribution of the thermal-neutron flux for determination of the fuel disadvantage factor.

Calculational methods[1–4] have been developed to make effective use of the experimental information. The methods provide the means of interpretation of the experimental results and permit interpolation or extrapolation (within limits) of the experimental data to reactor lattices not explicitly covered in the experimental investigations.

At the present stage of knowledge of basic reactor physics and constants, and with the presently developed calculational techniques, it is not possible to rely exclusively on theory for a mathematical description of lattice properties. Hence the approach is to resort to simplified semiempirical treatments which cover, in a gross sense, the broad features of the physical processes involved. Some measure of arbitrariness in the choice of physics constants cannot be avoided and, in general, a choice of constants is made which gives the best agreement of calculated and experimental results over a wide range of lattice configurations. With this approach, the value of the "adjusted" constant cannot be expected to have the exact physical significance represented by the conceptual constant.

The underlying basis for analysis of the experimental data used in references 1 to 4 is that it is possible to calculate, with adequate accuracy, the fast-fission factor ϵ, thermal-utilization f, and slowing-down and thermal-diffusion areas τ and L^2. On this basis, measurement of B_m^2 gives the product ηp, where η is the thermal regeneration factor and p is the resonance escape probability. The methods used to calculate ϵ, f, τ, and L^2 in references 1 to 4 are described in detail. Comparisons are given in references 1 to 3 of calculated and measured fast-fission ratios, R, from which ϵ is obtained directly; the calculational methods are similar to that given by Spinrad[6,7] (also in reference 4). From comparisons of measured and calculated disadvantage factors, one would conclude that the thermal utilization f is probably the most accurately known of the basic reactor constants.

A summary of calculated properties of 21 lattices measured in ZEEP and a comparison of calculated and measured bucklings B_m^2 are given in reference 1. The value of η was taken from Westcott's compilation[8] for $T = 20°C$, $r = 0.07$, as 1.326. The results of Swedish experiments[9] were used to derive semiempirical expressions for effective resonance integrals (RI) for either U-metal or UO_2 rods. The method used to calculate p follows closely that given by Critoph[7,10] and involves an adjustable parameter E_R which conceptually represents a single effective energy at which the U^{238} resonance capture occurs. The calculated lattice properties and the comparisons are made for an E_R of 25 ev and 7 ev. Reasonably good agreement of calculated and measured B_m^2 values is indicated; it is not apparent which value of E_R gives best over-all agreement.

The constant chosen for adjustment in the correlation of experimental data in reference 2 is η. The value of p was calculated by Critoph type formulation[10] using Hellstrand's[9] measured effective resonance integrals. The experimental data analyzed include those obtained at several laboratories, both in critical and exponential experiments. The average values, η, deduced from analysis of each group of measurements are tabulated in Table III-1.

The conclusions drawn from this analysis of 105 buckling measurements are:

Table III-1 VALUES OF η FOR VARIOUS TYPE FUEL ELEMENTS

Type fuel element	Measurement	Ref.	Measurements analyzed	η, average value	$\Delta\eta$, average absolute deviation
U rods	Exponential	11	22	1.316	0.004
U rods and rod clusters	Exponential	12, 13	32	1.320	0.004
U rods and rod clusters	Critical	10, 14, 15	27	1.303	0.003
UO_2 rod clusters	Exponential	5, 16	20	1.302	0.006
UO_2 rod clusters	Critical	17	4	1.290	0.003

1. With the calculational method and homogenization technique used for calculating the parameters for clustered fuel elements, the deduced η values are fairly independent of size and shape of fuel element and of lattice spacing in *each* group of measurements.

2. There is a difference in η for metal and oxide lattices indicating some systematic error in the theory, probably in the calculation of p.

3. There is a significant difference between η obtained from critical and exponential experiments. The reason for this discrepancy is not known.

The general philosophy used in reference 3 for correlation of their experimental data (250 different lattices) is as follows:

1. A fixed value of η equal to 1.327 is chosen throughout the calculations. This is the value of η obtained from use of Report BNL-325 for a purely Maxwellian thermal-neutron spectrum at $T = 20°C$. (Also, this value corresponds to the one given in Westcott's compilation.)[8]

2. The resonance escape probability p is related to the U^{238} effective resonance integral, RI (and to the pertinent geometrical characteristics of the lattices) by the method proposed by Critoph.[10]

3. The values of RI thus deduced from all the experimental data on metallic fuel are plotted as a function of S_{eff}/M using a value of $E_R = 30$ ev (where E_R is the previously defined effective energy). This E_R value was found to give minimum scatter of the data points. From a least square fitting to the data points, a curve of RI versus S_{eff}/M is obtained.

4. The RI values thus obtained are artificially defined. However, their use with $\eta = 1.327$ and with the calculational procedure adopted for ϵ, f, τ, and L^2 will give reasonably good agreement with the experimental results. Hence the "recipe" obtained can be used with confidence within the range of lattice configurations tested. It should be pointed out that, because of the artificial nature of the derived RI values, calculation of the initial conversion ratio from the derived RI values will be subject to uncertainty and will require experimental checking.

5. A similar procedure is followed for the oxide-fueled lattices, thereby obtaining a formula for RI that fits the oxide data.

The data were analyzed in reference 4 to lead to "consistent" values of η and RI. As a result of this analysis, the adopted values are

$$\eta = 1.294$$

and

$$RI = 2.13 + 9.16\sqrt{S/M}$$

for U metal. The reasons for the low value of η deduced from the analysis are felt to be the inadequacies of the relation between p and RI used and the inadequacy of the present form of the two-group and four-factor theory. The importance of evolving a satisfactory theory for p is emphasized. Comparisons are also given in reference 4 of the results of calculations and of experimental measurements by other investigators.

Reference 18 presents a survey of the available literature on measurements of the age of fission neutrons to indium resonance in heavy water. It was found that four separate measurements have been made between 1944 and 1956. The results

range between 104 and 111 cm^2 and are in close agreement considering the experimental uncertainties involved. The most recent measurements, discussed in reference 19, appear to be the most elaborate and reliable yet made. These measurements give a value of 109 ± 3 cm^2 for the age in 99.8 per cent D_2O.

Theoretical evaluations of the age show much closer agreement with the experimental results than is the case for light water. Although experimental measurements have shown that a Gaussian slowing-down model is not applicable to heavy water, it is found that numerical integration of $D/\xi\Sigma_s$ over the lethargy range from fission energy to indium resonance is in good agreement with experiment. Monte Carlo calculations, discussed in reference 19, give values near 110 cm^2 but are not yet fully corrected for the effect of anisotropic scattering.

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*Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, September 1958. (Individual papers available from the Office of Technical Services, Dept. of Commerce, Washington 25, D. C. Bound volumes of papers will be available from United Nations, Office of Public Information, New York.)

The experimental investigations of Core and Sato[1] have led to two empirical equations defining the burn-out limits of two organic coolants.

The first equation gives the critical heat flux, $(Q/A)_c$, for diphenyl and is based upon 48 test points.

$$(Q/A)_c = 454\ \Delta T_{sub}\ V^{0.63} + 116{,}000 \qquad (1)$$

where $(Q/A)_c$ = critical heat flux, Btu/(hr)(sq ft)
ΔT_{sub} = subcooling temperature (saturation temperature minus bulk-liquid temperature), °F
V = coolant velocity, ft/sec

The ranges of variables covered in the tests were: velocities of ½ to 17 ft/sec, pressures of 23 to 406 psia, subcooling of 0 to 328°F (also low quality tests), and bulk temperatures of 510 to 831°F. Accuracy is such that 92 per cent of the points fall within the ±2 standard errors. The standard error was found to be about 50,000 Btu/(hr)(sq ft).

The second equation defines the critical heat flux for Santowax R, the organic generally considered for use in reactor coolant and moderator applications. The equation is based upon ten test points.

$$(Q/A)_c = 552\ \Delta T_{sub}\ V^{2/3} + 152{,}000 \qquad (2)$$

where the symbols are identical with those defined above and the ranges of variables were: velocities of 5 to 15 ft/sec, a pressure of 100 psia, and bulk-liquid temperatures of 595 to 771°F. All data points fall within the ±2 standard errors, and one standard error was calculated to have a value of 80,000 Btu/(hr)(sq ft). The authors point out the limitations in extrapolating to higher pressures due to the inaccuracies in estimating Santowax R vapor pressures from the available low temperature data.

Critical heat-flux data were obtained for a polyphenyl mixture (15 per cent diphenyl, 58 per cent *o*-terphenyl, 24 per cent *m*-terphenyl, and 3 per cent *p*-terphenyl) but no simple correlation could be found. These data are reproduced as Fig. 3. Ranges of variables were: velocities of 5 to 15 ft/sec, pressures of 64 to 318 psia, subcooling temperatures of 9 to 383°F, and bulk-liquid temperatures of 565 to 811°F.

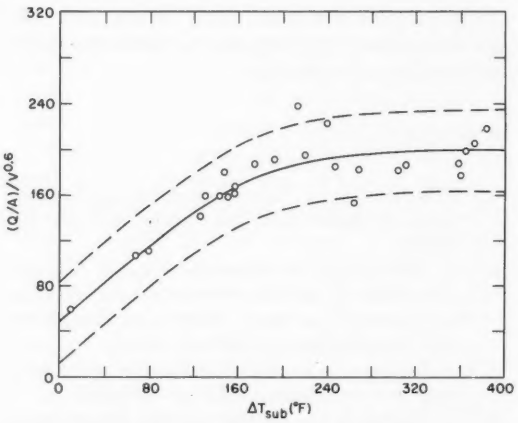

Figure 3—Critical heat flux for the mixtures of terphenyls and diphenyl as a function of velocity and subcooling showing lines at ±2 standard error. $(Q/A)V^{0.6}$ = (Btu/ft²-hr)/(ft/sec)$^{0.6}$.

Attempts to correlate critical heat-flux data from six tests on monoisopropylbiphenyl (MIPB) gave very poor results.

Since a single test section was used for all of the experiments, the effect of geometry as a variable was not considered. This limitation should be recognized when applying the correlations. The test section used consisted of an electrically heated tube, ½ in. in outside diameter (0.020-in. wall) with an unheated rod (0.188 in. in diameter) inserted coaxially to form an annular passage. Data points were obtained by adding heat in small increments until the onset of film boiling, characterized by a rapid rise in the outer wall temperature. The critical heat flux is defined as the last steady-state data point before film boiling occurs. The authors consider critical heat flux

to be a more consistent and reliable parameter than burn-out flux; these two points do not differ from each other by an amount which is significant in determining the heat-flux limitation.

The investigators performed five preliminary tests with water to check the apparatus and instrumentation. The results of these tests were compared with the correlation of Bernath[2] for water and showed close agreement. A comparison of critical heat fluxes expected in organically cooled reactors with those predicted in nonboiling water-cooled reactors at characteristic operating conditions for both types reveals a decided advantage for the water-cooled reactors of the order of several fold.

The authors mention that the burn-out and heat-transfer characteristics of the polyphenyls are "appreciably affected" by small concentrations of high vapor pressure impurities, such as water. If water were present to the extent of 0.1 wt.%, it was calculated that the saturation temperature of the polyphenyl mixture would be decreased by 90°F at 100 psia. Since burn-out is proportional to the degree of subcooling, this effect could be important. One point not discussed was the effect of radiation decomposition products on the burn-out heat fluxes.

References

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A number of papers have been published recently dealing with reactor safeguards. Most of those containing new technical developments are in the field of secondary containment for the reactor. The following discussion is oriented toward that consideration.

Basic Approaches to Safety and Safety Evaluation

The potential danger of a nuclear reactor to off-site personnel and property stems from the very large quantity of radioactive material which builds up in most reactor systems. As a rough rule of thumb it may be said that the fission-product radioactivity content ($\beta + \gamma$) of a reactor very shortly after shutdown is of the order of 10 curies per thermal watt of reactor power. For operating times of practical interest (hours or more), the activity level immediately after shutdown is not very sensitive to the duration of operation, but the rate of decay of the activity is sensitive to this variable. For reactors which have operated for a long time, the rate of decay is low. In a reactor which has operated for a year, a year of shutdown will reduce the activity less than a factor of 100 below that shortly after shutdown. When it is considered that the tolerance inside the human body for many fission products is measured in microcuries, the potential hazard from fission products, if they could escape from the reactor and become dispersed, is very large. An analysis of the possible effects of large-scale fission-product release led to the conclusion that if half the total fission products were released from a reactor of 500-Mw heat output, located in what might be an average location, about 30 miles from a major city, the total number of deaths might range from 0 to 3400, the number of injuries from 0 to 43,000, and the property damage from one-half million dollars to seven billion dollars.[1]

This potentiality of the reactor for widespread damage if uncontrolled dispersion of fission products could occur has required a very conservative approach to the question of reactor hazards. From the earliest days of reactor design, it was realized that every reasonable precaution should be taken to prevent accidents which might allow the escape of fission products from the reactor envelope. As soon as the problem of locating reactors in populated areas was faced, a further precaution was adopted, which consisted of enclosing the reactor in a secondary containment shell whenever it was deemed conceivable that an accident could occur which would allow the escape of fission products from the reactor proper. The secondary containment vessel was considered an extra safety precaution provided for the sole purpose of preventing widespread destruction and damage in the case of an accident of high destructive potential, which was considered to have very low probability of occurrence.

More recently, it would appear that these original considerations have been somewhat modified and that present safety criteria specify that *no* injuries should result to off-site personnel even as a result of the maximum credible accident. Thus the following two criteria are given by C. K. Beck for the reactor-site combination:

1. For normal operations the effluent streams (atmosphere, surface, and earth) must be such that radiation levels at the site boundaries and beyond do not exceed maximum permissible levels for continuous exposures.

2. In emergency situations the quantities of radioactivity that would be released beyond the site boundary, for even the worst accident the occurrence of which would be considered credible, must be such that possible dose to people would not exceed the permissible maximum emergency dose.[2] The permissible maximum emergency dose has not up to now been specified, but the indications are that it is unlikely to be larger than 25 rem.

It is of course generally recognized that the uncertainties in accident analysis and contain-

ment design are great enough that some uncertainty as to the possibility of fission-product dispersion remains even after the best possible containment design has been made. Thus C. R. McCullough, in discussing the containment of research reactors (and presumably the same conclusion would hold for power reactors), states: "It is worth commenting that even though great pains are taken to provide a steel safety container, which is tested to prove a low leakage, there is not a positive guarantee that such a container will protect the public from any and all accidents. In spite of all the foresight that can be used, the accident might be more violent than anticipated and might rupture the container, or leakage could have developed at the many penetrations through the container such as pipe, wires, etc., and finally someone might have carelessly left a door open, a ventilating pipe open, etc., at the time of the accident. For this reason location must always be considered, and it seems sensible to put the powerful testing reactors at locations as far removed as possible from populations and to use such reactors for the hazardous experiments."[3]

Beck, Mann, and Morris[4] state: "From this information (the hazards report), judgment must be made by the staff of the Commission as to the adequacy of the safeguards for the protection of the public. The judgment is a subjective one arrived at from a judicious weighing of the many complex features from which hazards might arise and the safety compromises that must be made in many aspects of design and operation. It might be supposed that criteria and standards could be devised against which each proposed facility could be measured. In due course, there may emerge some degree of generally acceptable standardization in reactor design, at least for certain particular types, but that situation is not yet possible. As long as so few reactors of similar design are built, and the cumulative experience is insufficient to give clear indication of those features which are in the long run most acceptable, the evaluative judgment of the adequacy of safety must remain a subjective one."

So long as subjective judgments are involved in the assessment of possible accidents, safety cannot be guaranteed by rigorous exposure criteria alone. There is, in fact, the possibility that the establishment of goals which are too difficult in this direction could lead to a relaxing of the subjective criteria. If the containment which must be provided for the maximum credible accident is too difficult, the designer is always faced with the strong temptation to decide that a somewhat lesser accident is the maximum credible one.

Actually, the requirement that exposure at the site boundary be limited to some relatively low value does not have any great effect on the specifications for integrity of containment vessels against the release of gaseous and particulate matter, at least so long as the vessel under consideration is the steel shell type which has been widely used on reactors of substantial power output. It has been found possible to fabricate these structures with very low normal leakage rates. If, on the other hand, it is assumed that an accident breaches the vessel, then the resulting leakage is usually too high by any rational standard. The one containment feature which has been affected by the criterion of low exposure at the site boundary is the shielding of the containment vessel. It has now become fairly common to provide extra shielding at the wall of the containment vessel, either just inside or just outside, to attenuate gamma radiation from fission products assumed to be dispersed inside the containment vessel by an accident. The cost of this extra shielding contributes significantly to the cost of the nuclear plant. In some cases reactor plant designers have attempted to minimize the cost contribution of this additional shielding by utilizing it as part of the normal biological shielding. This approach of course requires that the secondary containment vessel be uninhabited during reactor operations. In these cases the accident exposure criteria have quite obviously had a large effect on the basic approach to plant layout and operating procedure.

The cost of safety and secondary containment features is a subject of concern among most reactor designers. Recently Siddall,[5] of the Atomic Energy of Canada, Ltd., has questioned the rather large expenditures made for these purposes in the field of nuclear energy and has proposed certain new approaches to the question of reactor safety. He considers nuclear plants from two aspects, the humanitarian and the economic, and points out that both of these considerations would place a limit on the amount which should be expended in achieving safety of nuclear plants. From the humanitarian

point of view, Siddall postulates that the optimum degree of emphasis on reactor safety is that which results in the maximum life span for the average population. To the extent that nuclear plants have accidents which result in death or injury, they tend to decrease the average life span. To the extent that their output replaces that of other dangerous industries (e.g., the coal industry), they may effectively increase the average life span, and to the extent that they increase the general prosperity of the nation, they may also increase the average life span through the effect of better housing, better nutrition, better hygiene, etc., and through the progress of medical research. Siddall anticipates that the net effect of nuclear plants will be to increase the average life span; in the early stages this increase would result entirely from the greater safety of the nuclear industry in comparison to the coal industry. The humanitarian approach would be to find an optimum degree of emphasis to put on reactor safety, which would maximize this net increase in life span. The economic aspect of the problem appears straightforward in principle: The optimum degree of safety is that which minimizes the combined expenses of safety precautions and of damages from reactor accidents over a long period of time. Siddall estimates that the economic optimization requires somewhat higher safety standards than the humanitarian optimization but that our present preoccupation with safety is far greater than can be justified on either basis.

Siddall's thesis consitutes a logical approach to the abstract consideration of reactor safety, and no doubt it will color the thinking of many who are concerned with reactor safety. It does not, however, lead to a concept for the administration of reactor safety in a country like the United States, whose government is not one of planned humanitarianism, but a democracy. Beck, in commenting on Siddall's article, touches this point when he observes that there is a vast difference between the death of a worker by a recognized occupational hazard (which he has at least in principle accepted when he accepted his job) and the death of a member of the general public. It is this latter consideration, of death and injury to the general public which accounts for much of the expense of safety precautions, and which is the major concern of the reactor hazards group.

Practice in the United States

Perhaps the most obvious feature of safety practice in the United States is the widespread use of the secondary containment vessel. To a large extent this characteristic stems from the types of reactors used, yet it probably reflects also a special degree of concern with reactor safety. The necessity for the secondary containment shell has often been deplored as an obvious economic penalty on the nuclear power plant, and from time to time it has been suggested that the money invested in such containment might be spent more effectively in other ways to attain safety. But no other safety concept has yet been described which is so nearly independent of all other safety devices and safety features, and which can therefore give a comparable degree of safeguard against systematic errors in the safety analysis.

The secondary containment vessel is particularly attractive for the pressurized-water and boiling-water reactors developed in the United States. These reactors are compact and operate at relatively high power densities and specific powers. While these characteristics allow the use of reasonably small containment vessels, which are reasonably inexpensive, they also emphasize the desirability of a secondary containment structure, since the prevention of fuel melting in the case of a major break in the primary system is very difficult. These reactors generally contain rather large amounts of excess reactivity, controlled by rods of rather high individual reactivityworth; the possible rate of reactivity change by normal operating mechanisms is often high relative to that for certain other types of reactors. Thus a reactivity accident can be visualized by a somewhat simpler chain of errors than is necessary in the reactors having lower excess reactivity distributed over more elements. At the same time the water reactors are protected by strong negative temperature and/or steam void coefficients of reactivity. These in turn are not unmixed blessings since they contribute to the high excess reactivity of the clean cold reactor.

In the United States, the provision of a secondary containment shell is not considered to reduce the importance of other safety precautions but is considered a final barrier against the dispersion of fission products in case other

safety precautions prove inadequate despite careful attention. These shells are designed for leakage rates which would limit the escape of fission products to tolerable levels even if a large fraction of the fission products were released from the reactor proper. In the case of pressurized-water reactors, the vessels are usually designed to withstand internal pressures equal to those which would result from the sudden release of all the pressurized coolant in the primary system. Uncertainties are introduced into the design specifications by such questions as the necessity for protection of the vessel from internal missiles, and the possibility of explosive chemical reactions between fuel element or fuel jacket metal and water. In the Experimental Boiling Water Reactor (EBWR),[6] which uses metallic uranium fuel clad with zirconium, the containment vessel was designed to withstand an energy release equal to that resulting from a reaction of 25 per cent of the zirconium and uranium with water in addition to release of all primary water. Internal missile protection was provided in the form of blast shields and special design of the pressure vessel supports, as well as an internal concrete liner in the containment vessel.

The second type of power reactor which is being constructed in large size in the United States, the sodium-cooled fast reactor, also employs secondary containment. The possible sources of energy release which the containment vessel must be designed to withstand are nuclear releases and chemical energy releases from the sodium-oxygen reaction.[7] In the case of the Enrico Fermi reactor the processes determining the design were the burning of sodium (all the air in the containment building was assumed to react to the point where the oxygen content was reduced to 5 per cent) and the release of fission-product heat.[8] The EBR-II, although it is to be built on a remote site, is also being provided with secondary containment.[9] The considerations entering into the containment design are comparable to those for the Enrico Fermi reactor.

In Table V-1, reproduced from reference 3, the containment provisions for a number of U. S. research and power reactors are listed.*

*See also the table in reference 10, which includes silhouettes of the containment vessels for a number of reactors.

Where steel pressure vessels are provided as the containers, these have been built in accord with the pressure vessel codes of the American Society of Mechanical Engineers or of the American Petroleum Institute, whichever is applicable. There is every reason to believe that the actual leakage rates from many of these vessels are far less than the figures given in the table, which in general are limited by the means of measurement. As yet there has been no way fully worked out to perform a periodic testing of the tightness of these vessels on a practical basis, but this problem is under study. The method and procedure developed for the EBWR has been applied on two occasions with fair success; however, it requires shutdown of the plant for the duration of the test which may be a period of 24 hr or longer.

Despite the provision of containment vessels, it is the usual practice in U. S. safety analyses to consider the possible dispersion of fission products through air and ground water movements. This results from the previously mentioned circumstance that there is usually some uncertainty in the adequacy of the containment. The containment vessel must always have some opening, and it is conceivable that one of these might be open when an accident occurs. There are also uncertainties in the maximum accident which the containment vessel must withstand. In H_2O reactors these are usually connected with the conceivable occurrence of a major pressure vessel failure, or with the possibility of metal-water reactions. In fast reactors the firm limits to the possible nuclear energy release have not been established, although much work in this direction is under way. The basic features of fission-product dispersion by meteorological phenomena are outlined in the AEC publication "Meteorology and Atomic Energy."[11] Recent developments in this field have been summarized in reference 12, and reference 13 considers fission-product dispersion and related subjects. Gomberg, Bassett, and Velez[14] have made a quantitative statistical approach to the evaluation of the various factors affecting dispersion of fission products from a given site. Although the nature of the problem places large uncertainties on the estimates of results of any given accident, the approach does represent a method of evaluating various alternative sites relative to one another. Reference 15 is a brief discussion of considerations in the

Table V-1 SECONDARY CONTAINMENT PROVISIONS FOR U. S. REACTORS (FROM REFERENCES 3, 10, 29)

	Reactor Characteristics					Containment Shell Characteristics			
Reactor & date critical	Type	Power, Mw(t)	Fuel & jacket	Operating temp. & pressure	Gross vol. (cu ft)	Design pressure & test pressure (psig)	Plate material & thickness*	Specified leakage†	Shielding & exclusion radius
LPTR (1957)	Swimming pool (H_2O)	1	U-Al; aluminum	Ambient 15 psia	252,000	2		3	Shell only; 2400 ft
MITR (1958)	D_2O moderated & cooled, research	1	U-Al; aluminum	104°F 15 psia	190,000	1		1+	2 ft concrete inside, 110 ft
AFNETR (1959)	Tank	10	U-Al; aluminum	137°F	590,000	12.8 16	A201(A300) 0.563, 1 in.	0.00770	Shell only; 1500 ft
NACA-TR (1957)	Water moderated & cooled, pressurized tank, materials testing	60	U-Al; aluminum	187°F 135 psig	618,000	5	A285-54T 0.750 in.	0.008	Shell only; 3000 ft
GETR (1958)		33			230,000	5		0.4	Shell only; 2800 ft
WTR (1959)	Tank	20	U-Al; aluminum	Ambient 15 psia	260,000	2		1	Shell only; 2800 ft
APPR-1 (1957)	Pressurized water	10	UO_2; stainless	450°F 1200 psia	37,000	66.3 75	SA212 0.875 in.	0.0006+	2 ft concrete inside, 3 ft concrete outside; 1000 ft
PWR (1957)	Pressurized water	232	UO_2; Zircaloy-2	538.8°F 2000 psia	473,000	52.8 70	A201(A300) 0.75, 0.66, 1.25 in.‡	0.02	Concrete outside, underground; 2600 ft
YANKEE (1960)	Pressurized water	392	UO_2; stainless	550°F 2000 psia	1,020,000	34.5 43	A201(A300) 0.875, 1.25 in.	0.1	Shell only; 2000 ft
Con. Ed. of N. Y. (1960)	Pressurized water, Th converter	500	UO_2-ThO_2; Zircaloy-2	510°F 1500 psig	2,140,000	25 31.25	A201(A300) 0.89, 1.03 in.	0.1	Concrete outside 1600 ft
N. S. Savannah, Merchant Ship (1960)	Pressurized water	74		1750 psia		186			
EBWR (1956)	Boiling water	20	U-Zr; Zircaloy-2	488°F 600 psig	400,000	15 18.75	A201(A300) 0.375, 0.625 in.	0.0013+	1 ft concrete inside; 2000 ft

GE-BWR (1957)	Boiling water	50	UO_2; stainless	545°F 1000 psig	125,000	45 56	SA212 0.436, 0.875 in.	0.02+	Shell only; 2000 ft
Dresden (1959)	Dual cycle, boiling water	630	UO_2; Zircaloy-2	547°F 1000 psig	3,600,000	29.5 37	A201(A300) 1.25, 1.4 in.	0.5	Shell only; 2600 ft
Northern States Power Co. (1962)	Controlled recirculation, boiling water	203	UO_2-Al (boiling section); UO_2 dispersion in S.S. (superheater)	600 psia	Steel shell				
Elk River (1960)	Boiling water	73	UO_2-ThO_2; stainless	540°F 900 psig	405,200	21 26.25	A201(A300) 0.70, 0.50 in.	0.1	
S1G-SIR-A (1955)	Sub. intermed. reactor		U metal; stainless		5,460,000	20 25	A201(A300) 0.9, 1.1 in.	0.5	Shell only; 1 mile
SRE (1957)	Sodium cooled, graphite moderated	20	U metal; stainless	960°F 3 psig	Primary components are installed in sealed vaults within a tight building of special design.				
GGR (1961)	Sodium cooled, graphite moderated	245		18 psia	Containment system to be similar to SRE.				
EBR-II (1960)	Sodium cooled, fast breeder	62.5	U metal; stainless	900°F 100 psig	140,000	24 30	A201 0.5, 1 in.	0.04	14 in. concrete inside; 10 miles
ENRICO FERMI (1960)	Sodium cooled, fast breeder	268	U-Mo; stainless	800°F 92 psia	500,000	32 40	A201(A300) 1.03, 1.25 in.	0.025	Shell only; 2900 ft
OMRE (1957)	Organic moderated & cooled	16		300 psia	None. Reactor installed in cylindrical corrugated steel housing.				
Piqua (1961)	Organic moderated & cooled	45.5		140 psia	Primary components to be installed in sealed vaults.				
Plutonium Recycle Test (1960)	D_2O moderated & cooled	70	Pu-Al, UO_2; Zircaloy-2	1150 psia	504,500	15 18.75	A201(A300) 0.375, 0.625 in.	0.2	
HRE-2 (1957)	Homogeneous, D_2O moderated	5.2		2000 psia		30 32			

* Plate material, ASTM designations; designation in parenthesis is heat-treatment. Plate thickness: two numbers; first is thickness of heads; second is thickness of wall. Where only one number is given it is for wall.

† Mounted on top of reactor, containing sphere is 18 ft diameter × 20 ft cylinder with hemispherical top head that covers control-rod drive mechanisms; plate thickness is 1.25 in. plus 0.5-in. liner. Two smaller cylindrical vessels have 1 in. plate for heads nearest sphere.

‡ Leakage is expressed as per cent of volume (measured at STP) per psig. The value given is the design or guaranteed value unless marked + in which case it is the test value.

choice of reactor sites and, in particular, gives references for exposure and release limitations.

Practice in the United Kingdom

The British gas-cooled natural-uranium reactors, in contrast to the water-cooled types, are large in size and operate at low power densities and specific powers. They are not provided with secondary containment structures, and it is questionable whether such structures for these reactors would be economically practical. The basis upon which the gas-cooled reactors are judged to be acceptably safe appear to be the following:[16,17]

1. The excess reactivity is normally low and cannot be made very large so long as natural-uranium fuel is used. The reactivityworths of single fuel elements and single control rods are small. The normal rate of possible reactivity increase through control-rod withdrawal is very low, having, for example, an average value of $2 \times 10^{-6}\ k_{eff}$/sec for Calder Hall and $3 \times 10^{-6}\ k_{eff}$/sec for the Berkeley plant;[18] the latter rate corresponds to a requirement of 6 to 7 hr for bringing the reactor from complete shutdown to criticality and a total start-up time to power of some 11 hr. These slow rates of start-up are tolerable because shutdowns are rare, the nuclear plants being used for base load generation and the later plants being designed for reloading during operation. Because of the combined effects of these characteristics, the possibility of a rapid reactivity accident seems quite remote.

2. There is a relatively strong "prompt" negative temperature coefficient of reactivity, due to the Doppler broadening of U^{238} absorption resonances. This coefficient has the effect of slowing down still further any power increases which might be due to an accidentally high value of reactivity. It also tends to make the fuel-element temperature self-regulating. The gross power control in the Calder Hall reactor is effected simply by controlling the flow of primary coolant; slight movement of the control rods is necessary only for compensating certain second-order effects after a power change is made. Some of the benefits of the prompt negative temperature coefficient are negated, however, by the delayed moderator coefficient of reactivity, which results primarily from the effect of moderator temperature on effective neutron temperature, and which may become positive after a substantial quantity of plutonium has been formed in the fuel elements. The action of this coefficient is, however, quite slow because of the high heat capacity of the moderator and the low rate of heat generation in the moderator relative to the rate of generation in the fuel, and it is generally assumed that any autocatalytic effects originating in moderator temperature will be slow enough to be handled easily, even if they are not overcompensated by the negative fuel temperature coefficient.

3. It is practical to keep the normal fission-product content of the reactor coolant quite low because the rate of corrosion of the fuel by the coolant (in the event of a fuel can failure) is relatively low and because it is feasible to provide a good system for the detection and location of failed fuel elements. Consequently an important release of fission products from the reactor proper can occur only as the result of an accident which causes both a large-scale failure of fuel elements and a failure of the primary coolant system.

4. Because of the relatively low specific power, shutdown cooling of the fuel does not present a difficult problem. In the case of the Calder Hall reactors, auxiliary blowers are provided for circulation of the coolant in case of failure of the normal blowers, but natural circulation of the coolant is said to be adequate to cool the elements after the control rods have been inserted provided coolant pressure is maintained. If both coolant pressure and the circulating pumps are lost, the fuel will overheat, but it is expected that heat loss to the graphite will so delay the heating of the fuel elements that time would be available to assess the situation and take action to provide other cooling means before melting of the fuel cans occurred.

The question arises of whether, in the event of an accident which limits fuel cooling, the burning of fuel, fuel cans, or graphite, such as that which occurred in the Windscale accident,[19] could occur. British studies of the oxidation of reactor materials have shown that the oxidation of graphite by CO_2 is an endothermic reaction; that, although the fuel jackets (magnox) are oxidized by CO_2 at temperatures from 450°C to 640°C (the melting point), the oxidation is not rapid enough to represent a significant source of heat; and that the reaction between uranium

and CO_2 becomes self-sustaining above 650 to 700°C. Results of experiments in air show that the oxidation of graphite is negligible in the normal working temperature range and up to 400°C, that it becomes rapid enough to release 5 kw per channel at 500°C, and increases rapidly with higher temperature. The oxidation of magnesium in air was found not to be self-sustaining up to the melting point. The oxidation of uranium in air becomes significant at 150°C and self-sustaining above 350°C. These results indicate that the most serious accident might be one which would release CO_2 and let air into the system with no means for forced circulation. Such an accident would apparently require a major break in the pressure vessel, which is considered extremely unlikely. The possibility of such an accident is guarded against by designing the pressure vessels to operate at temperatures well above the brittle-ductile transition temperature, by rigid inspection of the vessel during construction, by periodic inspection of accessible points during the life of the vessel, and by the periodic observation of samples which are irradiated in the reactor under conditions similar to or more severe than those of the vessel itself.

It is not entirely clear what basic containment approach will be necessary if reactors of considerably higher specific power are developed through the use of enriched fuel. The proposed British Advanced Gas-cooled Reactor[20] is described with a secondary containment vessel. This feature, however, has been attributed to the experimental nature of the plant. In their studies of high-temperature gas-cooled reactors employing steel-jacketed fuel elements, neither Oak Ridge[21] nor the Kaiser-ACF group[22] postulated the use of a secondary containment vessel.

In the British design for the fast-neutron sodium-cooled reactor, at Dounreay, a secondary containment vessel is used.[23]

Other Countries

In a number of European countries, notably Sweden,[24] Norway,[25] and Switzerland,[26,27] nuclear reactors are being designed or built for process and space heating purposes as well as for the generation of electricity. Some of these reactors, although of relatively low power, are particularly sensitive from the point of view of hazards because they must be located near cities. The problem of containment for these reactors is generally being approached by locating the reactors in underground rock caverns. This type of containment is discussed further in a following section.

Published accounts of the approach to reactor containment in the USSR are not available. However, during one of the discussion periods at the 1958 Geneva Conference, F. A. Skvortsov made some comments on the subject. He stated that the practice is to insist that there be no dwellings within 1 km of a reactor site and no towns within 25 km. He indicated, however, that these considerations are appropriate only to the early stages of the reactor program and that they would probably be modified in the direction of moving the reactors closer to centers of population for reasons of economy. He mentioned the installation of the reactor and its primary circuit within a reinforced-concrete box capable of withstanding $3\frac{1}{2}$ atm pressure, with a steel covering to ensure gas tightness and automatic sprays which start automatically when the pressure in the box rises. He stated that reactor safety in the USSR is administered by the Ministry of Health.

Containment Structures

A number of papers have been published recently which relate directly to various aspects of secondary containment structures. Reference 28 is a bibliography on reactor containment. References 10 and 29 are discussions of the general subject of containment vessel engineering, which summarize the practice in secondary containment design in the United States. Chave and Balestracci[29] discuss the cost of providing such containment structures for nuclear plants. They estimate that the direct cost of the container itself, its foundation, lining, and associated equipment, will add about $15 per kilowatt of electrical capacity to the cost of a large nuclear power plant, above the cost which would apply if conventional plant housing were used. In addition, they point out that space limitations imposed by the container may increase the cost of plant components (e.g., vertical steam generators may be required instead of less expensive horizontal ones) and that the use of the container leads to less

efficient scheduling of the plant construction work. They estimate that when all these factors are considered the provision of secondary containment will increase the cost of a large nuclear plant by about $25 per kilowatt of electrical capacity; this extra construction cost is stated to represent a penalty of 0.5 to 0.8 mill per kilowatt hour on the cost of electrical power.

In a previous study by Kallman and Hanson,[30] the cost of reactor safety provisions was estimated for a pressurized-water reactor in the 500 Mw(t) range, the study being based on a reactor approximately like the Indian Point reactor of the Consolidated Edison Company. Items classified as special safety equipment included a steel containment sphere with its associated foundation and structural work, a cooling system and spray system for the sphere, a special ventilation system, and a sphere-pressure regulating system, as well as extra shielding around the sphere for gamma-ray attenuation from a hypothetical dispersion of fission products within the sphere. The estimated cost of these items, connected directly with secondary containment, totalled $5,000,000. Other special safety equipment considered included auxiliary power supplies, shutdown cooling and emergency poison injection systems, and instrumentation exclusively for safety purposes. A cost for extra transmission line was also included, on the assumption that safety considerations would indicate a reactor location 30 miles from that which otherwise would be the optimum. The total cost of all the special safety equipment was estimated at $8,400,000. On the assumption that conventional housing for the plant would cost $3,000,000, the net cost of the safety provisions was estimated at $5,400,000. This would represent a specific cost of $33 per electrical kilowatt for a plant having a nuclear electrical output equal to that of the Indian Point plant (163 Mw). It may be noted, however, that the estimated excess cost of the items directly connected with containment and shielding of the containment sphere amounted to only $2,000,000.

Other recent work has dealt with details of the specification and design of containment structures. In the case of a steel containment vessel, it is practical to make the vessel proper essentially airtight and to provide closures which have quite low leakage rates for the necessary openings in the vessel. The specification and design of the structure from the standpoint of leaktightness is therefore a relatively straightforward matter. The difficulties in arriving at design specifications have to do with the degree to which the vessel must be capable of withstanding internal pressures and other forces which may conceivably be connected with reactor accidents. Reference 31 is a discussion of the hydrodynamic aspects of this problem. For the analysis and discussion it is assumed that an explosive release of energy originates in the reactor core, and the methods of protecting the containment shell against the effects of this explosion are considered. For the general case it is assumed that the primary reactor vessel is breached, and that blast shields must be provided to protect the secondary containment structure from the localized effects of the blast, while the strength of the secondary shell itself is sufficient to contain the general rise of pressure which may result from the energy release. The propagation of the blast wave from the center of the explosion is discussed, as well as the mechanisms by which the energy in the wave is attenuated. The practical method of providing protection in the form of a blast shield is discussed. It is pointed out that the important mechanism of energy absorption in such a shield is through the crushing of materials in the shield by stresses which exceed their elastic limits. It is also pointed out that as the energy of the wave is dissipated materials of lower crushing strength are necessary to provide efficient energy absorption; it therefore is reasonable to construct such shields of materials of graded crushing strength, with the lower strength materials in the outer portions of the shield. The significance of the duration of energy release in the explosion process is discussed, and it is pointed out that, although very fast energy releases are more destructive near the point of explosion, slower releases may be more destructive at greater distances.

Reference 32 describes preliminary measurements of the dynamic behavior of porous materials such as those which might be used in a "sandwich type" blast shield comprising alternate layers of steel and compressible porous material (e.g., the EBWR blast shield). The report contains a discussion of the behavior of such shields and of the characteristics which are important to the energy attenuation properties of the porous material. These considera-

tions may be summarized in the following quotations:

> Pressure pulses are transmitted through structures as compression waves. The presence of porous media in a structure results in the modification of compression waves in three ways:
>
> (a) In a "sandwich structure" of a porous material and a dense, rigid material, the wave impedance mismatch between the materials results in reflections of the waves at interfaces, causing extremely rapid attenuation of shock fronts and of short, highly peaked compression waves.
>
> (b) The compression of most porous materials is largely irreversible so that a large fraction of the work done in compression then becomes unavailable as mechanical energy.
>
> (c) Where the relation between stress and strain is nonlinear, the form of a compression wave changes progressively. The nature of this change depends, of course, upon the nature of the nonlinearity of the stress-strain curve.
>
> The designer of a blast shield must consider all three of these effects. The magnitude of the first can be approximated in terms of the ratio of the wave impedance of the dense medium to that in the porous medium, the wave impedance being defined as the product of the density and the wave propagation velocity. Densities of materials are readily determined and wave propagation velocities for most of the dense media used have been measured. Propagation velocities of waves of the amplitudes of interest in the porous media are not in the literature.
>
> The second effect is related to the ratio between the energy expended in compressing the material and that recovered in its expansion. This latter may be evaluated either in terms of a measurable "coefficient of restitution" or as the difference between the integrals under the stress-strain curves in compression and expansion. To assess the stress patterns resulting from the last effect named, a knowledge of the stress-strain curve is, of course, required. Knowledge of this curve, in fact, would provide all of the other information needed. If the stress-strain curves were unrelated to time or strain-rate, a simple static test would yield all of the needed data. However, there is good reason to believe that the resistance of many porous media to compression is related to the rate of compression. Dynamic stress-strain data for the porous media to be used are necessary for reasonably optimum designs of blast shields. The present report describes an experimental technique which has been devised for the acquisition of such data.

From qualitative considerations of the above principles, the report states three rules-of-thumb for designers of blast shields:

> 1. In a sandwich structure a maximum attenuation of short shocks is obtained by the use of the greatest possible impedance mismatch. In practice, this implies the choice of a porous medium which has as low a density and shock velocity as is consistent with other aspects of design.
> 2. Maximum energy absorption results from the lowest possible "coefficient of restitution" or the largest possible difference between the stress-strain curve in compression and recovery.
> 3. Pressure "overshoot" is minimized by the use of materials whose stress-strain curves are convex upward. Continuous media with such characteristics are relatively rare; two notable examples are ice and mild steel. In porous materials the failure of the structure frequently results in effective upward convexity of the stress-strain curve. This same failure can result in very low coefficients of restitution.
>
> In selecting materials on the basis of these rules, consideration must be given to the fact that, for most materials, the properties involved vary with conditions.

The main body of the report describes the method for determining the dynamic behavior of the porous material and the preliminary results on samples of pine wood and insulating wallboard. The test samples are disks roughly 4 in. in diameter by $\frac{3}{4}$ in. thick. A test disk is sandwiched between a heavy rigid base plate and a thick upper plate of steel, the driving plate. The driving plate is driven suddenly downward by an explosive charge, and the motion of the plate and the compression of the sample are recorded continuously by a photographic "smear" camera. The compressive force on the sample can be derived as a function of time from the known mass and the observed deceleration of the driving plate. The dynamic stress-strain curves which result deviate significantly from the static stress-strain curves and show a marked dependence on rate of strain. These preliminary tests are sufficient to indicate the desirability of the dynamic data for blast shield design and to indicate the usefulness of the method for attaining such data.

Reference 33 discusses the technique of evaluating the integrity of a containment structure by means of scale model tests. It is pointed out that although designs based on purely theoretical approaches can produce safe structures such designs are apt to be overly conservative and the structures more costly than necessary. When tests are used to guide or to prove the design, the advantage of testing models of small scale is obvious. It can be

shown generally that valid scaling laws exist for the case in which only small deformations of elastic solids are involved; for the case of structures that undergo large elastic deformations or which deform plastically under transient loads, no general proof of the scaling laws has been developed, but the laws are inferred from theoretical treatments of a limited number of simple geometric cases. An experimental test of the validity of the scaling laws is currently under way at the Ballistic Research Laboratories on a series of homologous models of containment shells of different sizes.

Reference 33 also describes the tests of a one-quarter-scale model of the Air Force Nuclear Engineering Test Reactor which were carried out at the Aberdeen Proving Ground. The tests on the model simulated a series of reactor excursions increasing in severity from one which would produce insignificant damage to one which would damage the model so heavily it could not be economically repaired for further tests. Each test consisted of burning a propellant bomb or detonating a quantity of high explosive in the reactor-core section of the model. The total energy releases varied from 0.404 to 15.64 Mw-sec, which are computed to correspond to energy releases of from 26.0 to 1000 Mw-sec in the full-scale reactor. The tests indicated where minor changes were needed to make the full-scale reactor structure capable of containing the effects of a 1000 Mw-sec excursion.

References 34 and 35 describe experimental investigations which yield information on some of the details of the containment problem. In reference 34 the absorption of explosion energy by plastic deformation of steel containers of simple shapes was investigated and compared with theoretical predictions. The containers investigated were short lengths of steel pipe, 12 in. long, 5 in. in inside diameter and ¼-in. wall thickness, with massive end closures. The explosive energy was generated inside the containers by the burning of a propellant; the resulting pressure was transmitted to the walls of the cylinder through carbon tetrachloride which filled the space between the propellant charge and the cylinder walls. Initial tests were made with ordinary seamless steel tubing (MT-1015-20). Thirty-one grains (114.5 g) of type SPDN 6627-6″/47 solid propellant were sufficient to cause rupture of the tubing. The form of the rupture was that of a longitudinal slit extending the entire length of the tubing; the material of the tube underwent little plastic strain prior to rupture of the tube. Subsequent tests were made with five different tubes, all of which were cut from a single centrifugal casting of type 304 stainless steel which was specially cast for a high degree of homogeneity and isotropy. These test samples exhibited large plastic deformations before rupture, the maximum external radial deformation immediately prior to rupture amounting to 1.315 in. The rupture, when it occurred, had the nature of a rather general bursting rather than that of a split. These tubes, which were otherwise like the original seamless steel tubing, withstood the explosion of 72 grains (265.8 g) of the propellant without rupture.

Although these tests were primarily exploratory, and for the purpose of establishing experimental techniques, they do demonstrate the effectiveness of plastic deformation of vessel walls in absorbing the energy from an internal explosion. It is stated in the reference that it is known from a priori considerations that full-scale reactor pressure vessels will exhibit characteristics similar to those of the centrifugally cast tubing. The experimental program is being continued in order to yield basic containment design information for reactor vessels and dynamic structural response data useful in the design of blast shields.

Reference 35 describes experiments which investigate the penetration of steel plate by projectiles, such as those which might be postulated to originate inside the reactor containment structure during a reactor accident. The experiments described investigated projectiles having the form of long rods. It was found that a rather simple relation would express the penetrating ability of such rods over a rather wide range of rod size and plate thickness. The relation was of the form

$$e = 16{,}000T^2 + 1500T$$

where e is the kinetic energy per unit diameter of rod, in foot pounds per inch, and T is the thickness of mild steel plate, in inches. The formula applies specifically when the plate is supported by a rigid backing with a 4-in. opening and when the plate thickness lies between 0.030 and 0.25 in. The projectiles were of drill-rod steel with masses between 1 and

10 oz, diameters between $^3/_{16}$ and $^1/_2$ in., and velocities up to 400 ft/sec. In supplementary experiments the applicability of the formula was extended, by correction factors, to a rather wide range of other conditions. These experiments should prove useful in estimating the probability of puncture of the containment vessel by missiles and in designing shields against such missiles.

Reference 36 describes a method for evaluating the leakage from semiairtight buildings such as those that might be used on research reactors of relatively low power. The method was used to evaluate the containment effectiveness of the building for the Ford Nuclear reactor* at the University of Michigan. The method is based on a theoretical model which assumes that the leakage is due to bulk mass transport of gas through existing holes, rather than to gaseous diffusion. The leakage holes are treated as orifices, across which pressure drops can be established by two possible means: fluctuation in the ambient atmospheric pressure, and dynamic pressure differences due to wind velocity; the possibility of internal pressure generated by a reactor accident is not treated. The tightness of the building is specified in terms of a building leakage coefficient, which may be evaluated experimentally for a given building by pumping up the internal pressure above atmospheric and observing the rate of pressure leak-off. Having determined the coefficient, one can make a theoretical prediction for the probability of a given leakage rate if a history of meteorological behavior is available. Some uncertainty remains, however, which depends upon the distribution of the leaks. The most pessimistic assumption is that all leakage occurs through two holes, one of which is situated at the point of lowest dynamic pressure and the other at the point of highest dynamic pressure. A more realistic assumption is probably that the leakage is from many small holes uniformly distributed over the building surface. In the case of the Ford reactor, the assumption of uniformly distributed leaks gave the result that there is less than 2 or 3 per cent probability that a leakage rate greater than 5 per cent of the building volume per day would occur. The assumption of leakage concentrated in two large holes predicted a probability of about 20 per cent for a leakage rate as high as 15 per cent of the building volume per day.

*A pool type, H_2O-cooled H_2O-moderated research reactor of 1 Mw(t) output.

Underground Location of Reactors

The possibility of providing containment by the underground location of nuclear plants has received some attention in the United States,[37] and such installations are definitely planned in some other countries. In Sweden the Research Reactor R-1, which went into operation in 1954, is located in an underground chamber excavated in rock, and underground locations are planned for two district heating reactors which are now under construction.[24,38] The Heavy Water Boiling Reactor at Halden, Norway, is also being constructed in an underground rock chamber,[25] and a design for an underground installation in Switzerland[26,27] has been made.

A deeply excavated rock chamber obviously provides a reactor plant housing which can withstand quite energetic reactor accidents without large-scale failure. However, it is generally recognized that the simple excavation of a chamber in underground rock does not provide adequate leaktightness, and it is assumed that some sort of gastight liner is necessary, both for the prevention of gas leakage and for the prevention of ground water contamination. The problem of constructing such a gastight liner, and of maintaining the tightness over a period of years, may be a difficult one, particularly since there is no obvious simple way of locating whatever leaks may exist in the liner.

Other advantages and disadvantages of underground construction are discussed in references 25, 37, and 38. Reference 37 points out that, although there are a relatively large number of locations in the United States where underground plants could be built, there are additional site requirements, such as the availability of adequate water for waste heat removal, which must be met. These requirements may greatly decrease the number of suitable underground sites. The reference estimates that in the United States the cost of underground installation will be greater than the cost of a contained surface installation by perhaps 3 to 7 per cent of the total plant cost. It is estimated in reference 25, however, that in Norway the cost of an underground enclosure is comparable with the cost of a surface building

of ordinary concrete masonry construction and is considerably less than the cost of a steel containment vessel. Reference 38 estimates that in Sweden the underground building is economically competitive with a surface building provided a welded gastight steel lining is not required for the underground chamber.

Underground installations may of course be considered in order to give protection against external hazards, such as enemy action in time of war. In such cases this type of location has obvious advantages which have not been considered in the discussion above.

Other reports that may be found useful in safety and containment studies are listed in references 39 to 43.

Reactor Fuses

For a number of years schemes have been considered for the prevention of reactivity accidents through the use of nuclear fuses—elements of some type which would be installed as part of the reactor and which would have the characteristic of producing a large decrease in reactivity whenever the power level exceeded some predetermined safe value.

To achieve the desired reliability, the reactor fuse should involve the minimum possible number of mechanical or electrical linkages. It must, in this respect, make a substantial improvement over the normal control and safety system of the reactor; otherwise it can hardly be considered a fuse, but rather an additional safety feature in the normal control system. The fuse must also act rapidly. Evidently the degree of difficulty in devising and developing a practical fuse will depend upon the precision required in triggering the fuse. This precision will vary greatly with the type and application of the reactor. For low-power training reactors, in which the normal operating power is far below the dangerous level, the development of a fuse is relatively straightforward. It is also a reasonably straightforward development for research reactors which operate at relatively low powers. In high-power research and testing reactors, and in power reactors, the normal operating power is not far below that which will damage the fuel elements by overheating. In such cases the setting of the fuse trigger must be quite precise, and the precision must be attained despite any changes in power and neutron flux distributions which may occur during the life of the reactor. In the case of power reactors, the situation is somewhat more difficult because a relatively wide range of coolant temperatures may have to be accommodated.

The first transient tests of fuses in a reactor were made[44,45] on fuses constructed by General Electric, and by J. T. Weills of Argonne National Laboratory, with the BORAX-I reactor, in 1954. The Argonne fuse consisted of a cadmium cylindrical shell, spring-propelled into the reactor core when a solder restraining link was melted by the heating of a uranium strip. The fuse was found to be easy to use and limited the power of reactor excursions, which were made at periods as short as about 25 msec.

The components of the General Electric fuses were a pressure chamber filled with helium, a tube containing powdered normal boron, a fusible disk containing U^{235}, and a partially evacuated boron receiver chamber that surrounded a specially constructed fuel assembly. The pressure chamber, boron holder, and the fusible disk were located below the receiver chamber. The fusible disk, which was made up of lead-bismuth eutectic (Cerrotru) mixed with 31 wt.% UO_2, was ruptured by nuclear heating, allowing the compressed helium to force the powdered boron into the $\frac{1}{16}$-in. annular space surrounding the fuel assembly.

The fuse fuel assembly was made of MTR type fuel plates wound into spirals and stacked so that eight plates, which together contained 50.3 g of U^{235}, constituted an assembly. Four reactor fuses were placed in regular fuel channels in the BORAX reactor during the tests. Tests were made to measure the effect of the fuses on induced reactor power excursions. In the fastest test the reactor was subjected to an excess reactivity of 1.30 per cent k_{eff} which produced an exponential period of 14.6 msec. The four fuses removed more than 2.51 per cent k_{eff} and reduced the total energy of the excursion to 52 per cent of that which would have characterized the excursion in the absence of the fuses. Development work on means of propelling the absorbing powder into the reactor core is described in reference 46.

More recently, Atomics International Division of North American Aviation has been carrying out development work on fuses of various types for the Atomic Energy Commission.[47–50] Fuses for research reactors, power reactors, and

for the Hanford production reactors have been under development. Tests of two research reactor fuse types have been made under reactor transient conditions in the KEWB and SPERT reactors. The first of these fuses tested[51] was an electronically actuated one and thus did not satisfy all the requirements of self-sufficiency which one would desire in a fuse. The fuse consisted of a long, empty cylindrical chamber (the receiver chamber), which was installed in the core of the reactor, and an attached storage chamber which contained BF_3 under pressure and was installed outside the reactor core. The storage chamber was separated from the receiver chamber by a diaphragm which could be ruptured by a chemical explosive when the explosive was ignited electronically. The electronic signal originated in an ordinary ion chamber and was converted to a firing impulse by means of a thyratron circuit which was arranged to fail safe in case of loss of electric power. The device was tested in the KEWB-I reactor. In the fastest test the reactor was subjected to an excess reactivity which produced an exponential period of 160 msec. The fuse limited the power rise to 1.1 kw, 0.14 per cent of the peak value that would have been expected without the fuse. The movement of the neutron-absorbing BF_3 into the receiver tube decreased the reactivity of the reactor by 1.5 per cent k_{eff}. The time interval between trip of the device and insertion of the BF_3 was measured to be 4.2 msec.

The second fuse type tested was completely self-contained. It also utilized BF_3 stored under pressure, but the storage chamber was separated from the receiver chamber by a small intermediate chamber which was pressurized to an intermediate pressure between that of the storage chamber and that of the receiver chamber. The intermediate chamber was separated from each of the other chambers by rupture diaphragms. The intermediate chamber communicated with the receiver chambers by a tube which was normally plugged by a fusible solder containing U^{235}. At the trip level of power, the heat generation in the U^{235} melted the solder, allowing the gas to escape from the intermediate chamber, thus causing the rupture of the two diaphragms successively and the rapid release of BF_3 into the receiver chamber. Several tests of these fuses have been run[52] in the SPERT reactor, the most drastic at a reactor period of approximately 10 msec. The fuse terminated the power rise at a level 25 times lower than would have occurred without the fuse and limited the total energy generation to a value less than one-tenth that which would have been generated.

The work on fuses for the Hanford reactors is described in reference 53. Fuses have been constructed which are identical in size and shape with regular Hanford slugs and could be inserted in place of certain slugs. A large number of the fuse slugs would be required and they would cause some loss of reactivity in the unfired condition. Such fuses have not been installed in the Hanford reactors for routine operation.

One of the most important components of a reactor fuse is the triggering mechanism. For this reason General Electric conducted fuse trigger tests at the SPERT facility to explore the possible development of a more sensitive, faster acting trigger primarily for detonating cartridges that contain gunpowder as the propellant.[54] The tests dealt mostly with the use of bimetallic sensor strips which make electrical contact when subjected to nuclear heating. For comparison a fusible disk similar to those used in the BORAX fuse test was also tested. Several bimetallic sensors of various geometries were tested. One design referred to as a snap beam bimetal assembly operated very satisfactorily. It consisted of a strip of bimetal held in a bowed position with uranium on the concave side and zirconium on the other. During a reactor power excursion the increased fission rate in the uranium metal caused the bimetal strip to straighten and finally snap into a bowed shape opposite to its original position, at the same time making an electrical contact. This type of trigger was found to be much faster than the fusible disk but requires some design changes to keep it from acting when the ambient temperature changes without sacrificing its ability to limit a slow runaway of the reactor. This type of fuse trigger still has the disadvantage of not being self-contained (i.e., independent of outside power).

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*Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, September 1958. (Individual papers available from the Office of Technical Services, Dept. of Commerce, Washington 25, D. C. Bound volumes of papers will be available from United Nations, Office of Public Information, New York.)

The International Atomic Energy Agency has recently released publications dealing with the safe handling of radioactive material.[1,2] These are the work of an international panel composed of experts from various countries and are advisory rather than regulatory. The references contain considerable valuable data. The subjects covered are as follows:

1. Maximum permissible concentrations in air and water.
2. Maximum permissible surface contaminations.
3. Maximum permissible exposures to radiation for workers in the atomic energy industry and the population in general.
4. Storage and operations with sealed and unsealed sources.
5. Transportation of radioactive material.
6. Accidents.
7. Decontamination, waste control, and disposal.
8. Monitoring, medical supervision, and records.

An important consideration to the nuclear power industry is to avoid the creation of public hazard, and of major interest, of course, is the exposure of people to radioactivity. Reference 1 considers the whole population as divided into categories subject to three types of exposure.

1. Occupational exposure—exposure of an individual normally working in a controlled area wherein radioactive material is handled, stored, etc.
2. Exposure of special groups—people working or residing in proximity to a controlled area.
3. Exposure of the population at large.

The nuclear engineer is interested in such things as allowable concentrations in the waste discharges from nuclear plants, and some space will be devoted to the discussion of the maximum permissible genetic dose to the nuclear plant employees and the population at large. In 1954 the International Commission for Radiological Protection (ICRP) recommended a value of 0.3 rem for the maximum permissible weekly dose for *occupational* exposure, and the following fission-product maximum permissible concentration values were calculated on the assumption that under continuous exposure to the concentrations shown this value of dose would not be surpassed in the whole body or any critical organ.

Isotope	Dispersed in water, μc/cc	Dispersed in air, μc/cc
Mixed fission products	10^{-7}	10^{-9}

Reference 1 recommends that the permissible total dose, for occupational exposure, accumulated in sensitive organs now be computed by the following formula:

$$D = 5(N-18) \qquad (1)$$

where D is dose in rem and N is age in years. For a worker employed at 18 years of age, this implies a maximum dose rate of 0.1 rem per week for a working year of 50 weeks with an accumulated dose at age 30 of 60 rem. The maximum permissible genetic dose to the population as a whole has not been recommended, but it is suggested that the dose in addition to background and medical exposure not exceed 5 rem. The permissible genetic dose is defined as "that dose, which, if it were received by each person during the reproductive period, would result in an acceptable burden to the whole population." It is further suggested that the maximum permissible genetic dose be apportioned as follows:

	Per capita of total population, rem
Occupational exposure	1.0
Exposure of special groups	0.5
Exposure of the population at large	2.0
Reserve	1.5

The occupational exposure dose means that a small per cent of the whole population could accumulate the relatively large doses given by Eq. 1. The special groups could accumulate 0.5 rem with the population at large accumulating 2.0 rem. Including the reserve of 1.5 would result in the whole population receiving an average dose of 5.0 rem. An important consequence of these suggestions is that the average fission-product concentrations in public air and water supplies must be lower by a factor of 100 than the occupational concentrations given in the preceding paragraph.

Reference 2 considers the transportation of radioactive materials and is of interest primarily to show trends, since in the United States the Interstate Commerce Commission (ICC) as well as various state regulations govern shipments about the country. Unfortunately the draft has adopted the terminology of that section of the ICC regulations pertaining to fogging of photographic film, namely, "one unit of gamma radiation is that amount of gamma radiation which will have the same effect on sensitive photographic film as one milliroentgen per hour of the gamma radiation of radium filtered through 0.5 inches of lead. . . ." It would seem that a more useful standard could have been chosen in the form of number flux and spectrum.

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The design and the attainable performance of any solid fuel reactor are largely determined by the performance of the fuel elements which are available for use in the particular reactor type. Some of the fuel characteristics which have important effects on reactor performance were discussed briefly in the first issue of the Review (Vol. 1, No. 1, December 1957), and progress on fuel elements of various types has been reviewed in later issues. Another of the Quarterly Technical Progress Reviews, *Reactor Core Materials*, gives very detailed coverage to fuel element and fuel material development. No such detailed review is to be attempted in *Power Reactor Technology*, but developments which may have large effects on reactor design and performance will be noted, and from time to time fairly comprehensive articles will be written to brief the reactor designer on the status of specific fuel-element types. The articles below cover two fuel-element types which are not now in use in the United States but which may have promise.

Graphite–Uranium Fuel Elements

The Pebble Bed Reactor[1] concept makes use of fuel elements which consist essentially of spherical graphite elements containing intermixed fissionable and/or fertile material as oxides or carbides. Such elements have been under study for over 10 years but have only recently received a major development effort. It is hoped that a successful development will yield a cheap, simple element capable of heating the coolant gas to temperatures well above 1000°F. In the case of the 125 Mw(e) Pebble Bed Reactor (Sec. IX of this Review), proposed by Sanderson and Porter, helium is to be used as the coolant and the reactor outlet temperature of this gas will be 1250°F; the gas outlet temperature from the core is 1540°F and from the blanket 561°F. The reactor design calls for 1½-in.-diameter fueled graphite spheres; 228,600 core spheres of 90 wt.% graphite, 9.17 wt.% ThO_2, and 0.83 wt.% UO_2; 296,250 blanket spheres of 50 wt.% graphite, 50 wt.% ThO_2. To achieve the required core outlet temperature, the maximum surface temperature of the fuel is calculated as 2170°F and the maximum central temperature as 2440°F (references 1 and 2).

These fuel elements are to be operated without a metallic clad, and therefore the effects of temperature and irradiation on the elements become paramount in the design of the reactor. Dimensional stability must be good to enable foolproof refueling; fission product and fuel losses should be minimized to prevent excessive activity in the coolant loop; no welding of fuel elements to form large clinkers can be tolerated since this would affect the loading and unloading of the reactor and might tend to form a structure which, in shifting and breaking, might cause unexpected reactivity changes.

Because of the importance of these fuel elements to the feasibility of the Pebble Bed Reactor, Sanderson and Porter, under an AEC contract, have conducted a survey of the data available on fueled graphite elements.[3] It may be said, however, that if a high-temperature fuel element of low parasitic absorption, capable of containing fission products, can be developed, its applications will be much broader than as a fuel for a pebble bed reactor. The Sanderson and Porter survey and other pertinent data are summarized briefly below. A detailed bibliography of references is included in the Sanderson and Porter report.

Fueled graphite elements are made by:

1. *Impregnation* of a graphite shape with a uranium compound, followed by a conversion of the compound to the oxide or carbide by heating.

2. *Admixture*, or the mixing of a uranium compound such as the oxide with a graphite flour and a binder, followed by forming and baking to carbonize or graphitize the binder.

3. *Lumping*, where a lump or pressed shape of uranium oxide or carbide is sealed within a preformed graphite body.

Some of the temperature-dependent physical and chemical changes experienced by the ma-

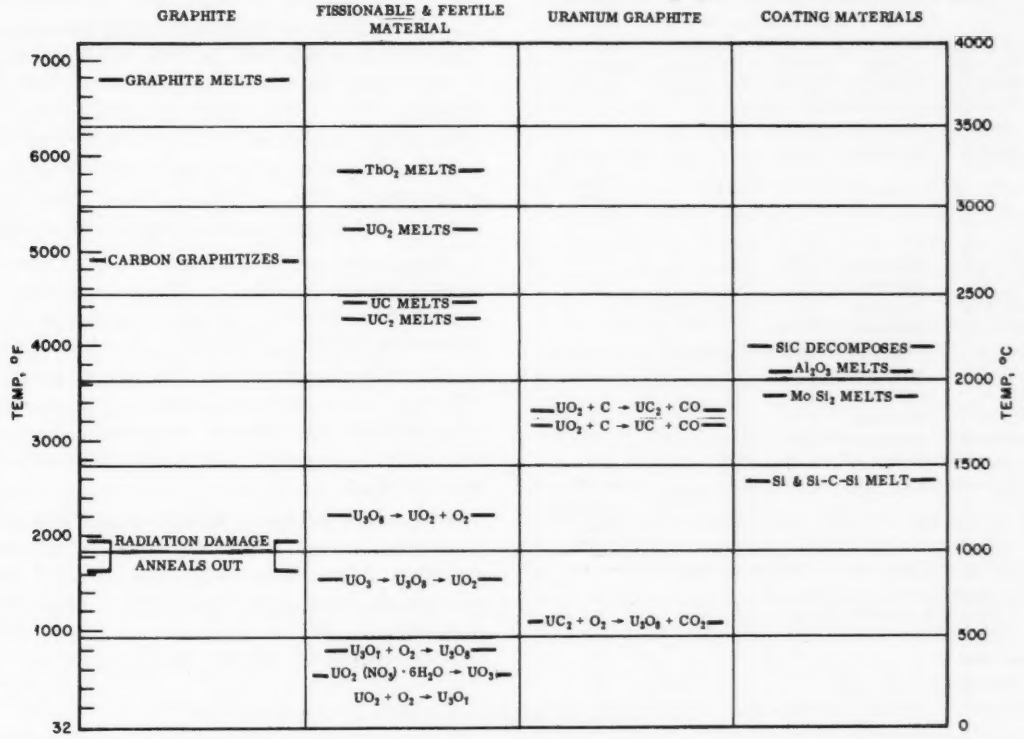

Figure 4 — Effect of temperature on pebble bed fuel reactor fuel materials.[3]

terials used in this type of fuel element are shown in Fig. 4.

The manufacture of fueled graphite bodies by impregnation techniques has been studied by Argonne National Laboratory (ANL), Atomics International, and Los Alamos Scientific Laboratory (LASL). Density and fuel loading data of the LASL graphite impregnations for Sanderson and Porter are shown in Table VII-1.

Fabrication of the fuel elements by admixture techniques has been carried out by Battelle Memorial Institute (BMI), National Carbon Company, Sylcor, and LASL. Some of the physical property data on an admixture type fuel element prepared by the National Carbon Company are shown in Table VII-2.

Sylcor, under an AEC contract, has also prepared admixture type fuel elements for the 125 Mw(e) Pebble Bed Reactor concept. One-inch blanket balls, containing 50 wt.% ThO_2, with a density of 2.802 to 2.823 g/cm^3 have been satisfactorily made.

Minnesota Mining and Manufacturing Company is investigating lumped type fuel elements to which their 3M "Ceramic S" coating has been applied to reduce the escape of gaseous fission products. Fully enriched UO_2 elements fabricated with this coating were leaktight to helium at room temperature and are now undergoing irradiation tests in the MTR. This company is also fabricating UO_2 lumped fuel elements, coated and uncoated, for study and irradiation at Oak Ridge.

Although the fabrication techniques of this type fuel have been developed extensively, very little is known of its irradiation behavior. Some of the experimental data available and reported in the reference report[3] are as follows:

1. *Impact Strength.* Gravity drop tests on machined and molded unirradiated balls indicate free falls of 98 to 238 ft are possible without fracture.

2. *Self Welding.* It is not considered likely that welding of graphite balls will occur since temperatures of 4000°C and pressures of 50,000

Table VII-1 RESULTS OF GRAPHITE–URANIUM IMPREGNATIONS AT LASL*

Type and sample No.	Mfg. method	Density, g/cm³	Wt.% UO_2, 1st impreg.	Radiographic analysis
AGOT-1	Machined	1.70	10.6	
AGOT-2	Machined	1.70	10.0	Edge, 171 mg/cm³; center, 140 mg/cm³
AGOT-3	Machined	1.70	9.2	
AGOT-4	Machined	1.70	9.5	
AGOT-5	Machined	1.70	10.5 (12.3)†	
AUC-1‡	Machined	1.74	3.2	
AUC-2	Machined	1.74	4.0 (6.1)†	
AUC-3	Machined	1.74	5.5	
AUC-4	Machined	1.74	7.0	
AUC-5	Machined	1.74	11.4	
AGOT-M-1	Molded	1.66	0.55	
AGOT-M-2	Molded	1.66	3.00	
AGOT-M-3	Molded	1.66	2.90	Edge, 95 mg/cm³; center, near zero

* Data taken from reference 3.

† Total after second impregnation.

‡ AUC samples showed variation of loading with impregnation time.

fuel element which contains fuel material and has no cladding or nonfuel-bearing coating. In experiments conducted in 1948, ANL reported a loss (from specimens of graphite of ¾ in. in diameter by 1½ in. long, of density 1.37 to 1.43 g/cm³, fueled with U_3O_8) of 0.86×10^{-3} to 2.6×10^{-3} μg/cm² hr at 1500 to 2000°F sample temperature. The cylinders were induction-heated and subjected to a flowing helium stream.

Battelle Memorial Institute reported results on UC_2 rods coextruded in unfueled graphite and subjected to flowing nitrogen or argon at temperatures of 2000 to 2500°F. Uranium losses of 6.2 μg/cm², dropping to 0.5 μg/cm² hr after 100 hr, were observed. Similar rods using UO_2 as the fuel showed a uranium loss of 0.5 μg/cm² hr, which dropped essentially to zero in 48 hr.

4. *Thermal Stress.* Nonirradiated graphite specimens have been subjected to thermal stresses of calculated magnitude greater than the ultimate strength of the material at the test temperature, but no failures occurred. This was attributed to plastic deformation of the material. Rapidly induced thermal stresses (thermal shock), such as might be caused in a reactor by sudden large power changes, may impose more severe conditions. Since thermal conductivity, fracture strength, and yield properties may be changed by radiation, radiation

Table VII-2 PROPERTIES OF MOLDED URANIUM–GRAPHITE BODIES*

Wt.% U	Bulk density, g/cm³	Thermal conductivity, Btu/(hr)(ft)(°F)	Modulus of rupture, psi		Compressive strength, psi
		A. Baking temperature, 1450°C; carbonized binder and uranium oxide as fuel			
0	1.72	12.7	6600		15,800
3.8	1.75	11.5	5470		18,600
11.7	1.88	9.8	5070		18,990
19.9	1.93	10.4	4290		14,100
		B. Baking temperature, 2800°C; graphitized binder and uranium carbide as fuel			
			Modulus of rupture, psi		
			With grain	Against grain	
0	1.70	26.0	6000	4680	10,070
3.9	1.78	26.0	5140	3800	9,060
11.7	1.89	32.0	4470	3400	8,300
20.8	1.99	33.0	3440	2060	5,570
23.7	2.09	37.0		2280	5,520
30.3	2.22		4230		5,940

* Data taken from reference 3.

psi are required to weld graphite under controlled conditions. Welding due to carbon transport or uranium migration is also considered unlikely.

3. *Fuel Loss.* Scattered data indicate there is some loss of uranium from the surface of a

effects must be known in order to predict the failure point of these fuel elements.

5. *Surface Stability.* There appears to be no indication that the coolant will cause erosion of the fuel elements once all the loose graphite has been removed from the surface. Test data by North American Aviation (NAA), on a graphite specimen heated to 2000°C and subjected to helium flowing through a nozzle at a throat velocity of 9000 ft/sec and 2000°C, indicated that an initial erosion of the loosely held particles, smeared into the surface by machining, took place for about 24 hr. After this initial period no further erosion was apparent.

Minnesota Mining and Manufacturing Company also conducted erosion tests on graphite capsules coated with SiC by their proprietory process.[4] In their tests a near-sonic jet of air was directed at a small area of the specimen for 1000 hr at 1000°C without evidence of damage to the sample and with a weight gain of only 0.1 per cent. Other tests were conducted with three cylindrical specimens heated to 1100°C and subjected to air at a velocity of Mach 0.25. Two of the specimens showed no evidence of failure. The third specimen started to lose weight at the end of 30 hr, and only a hollow shell of Si-SiC was left after 400 hr. Since the coolant was air, failure of the Si-SiC coating allowed air oxidation of the remaining graphite. No explanation of the specimen failure was advanced.

6. *Irradiation History.* A summary of irradiations of uranium–graphite fuel specimens was prepared by Sanderson and Porter and is reproduced here as Table VII-3.

These data are not sufficient to evaluate the fuel element for the Pebble Bed Reactor, but some results are of interest.

In none of the irradiations was there any report of crumbling or disintegration of the fuel elements under irradiation.

Thermal conductivity decreases by factors of 15 to 30 at low temperature (item 12, Table VII-3), 39 at 775°C (item 4, Table VII-3), and 2.1 to 1.4 at 1000 to 1300°C (item 13, Table VII-3) occurred with irradiation. The higher temperature irradiations appear to anneal fission fragment damage and hence give smaller changes of thermal conductivity.

Although the effect on thermal conductivity is important, other irradiation effects on the element properties such as tensile strength and plastic deformation would be desirable. No data appear to be presently available on these effects.

In a more recent report by BMI[5] data were presented on the irradiation stability behavior of specimens of fueled graphite prepared by three different manufacturers. Table VII-4 is a brief summary of the data obtained. Except for the small variation of diameter the fuel elements were reported to be dimensionally stable. Of the six specimens, one (BMI No. 11) had a crack in the graphite clad, indicating that some type of failure took place in or out of pile.

7. *Fission-product Leakage.* Since the fuel elements proposed for the Pebble Bed Reactor are to be unclad, the fission-product leakage may cause a severe radiation problem throughout the primary coolant system.

In an investigation by Doyle (item 9, Table VII-3), a graphite specimen impregnated with UO_2 was irradiated at low temperature and then allowed to cool out of pile until the short-lived precursors had decayed. The specimen was then heated and the remaining activities of various elements were measured. Table VII-5 summarizes these results.

These data would indicate that at high operating temperature most of the fission products with boiling points below the operating temperature would be released.

Sanderson and Porter, in their study of the feasibility of the Pebble Bed Reactor, assumed that all the fission products which are volatile below 2500°F would be released to the gas stream. The main contributors to radioactivity would be iodine, xenon, krypton, and bromine, and the activity level of these elements, excluding Xe^{135}, were calculated to be as shown in Table VII-6.

Under these conditions, approximately 300,000 curies/Mw(t) or a total of $\sim 10^8$ curies of activity would exist during operation of the reactor at 350 Mw(t), requiring heavy shielding of the primary loop, including the heat exchangers. Cottrell, in a paper on the release of activity from GCR systems,[6] calculated that the total fission-product activity released under the above assumptions would be about 7.3×10^5 curies/Mw(t) or 2.5×10^8 curies of activity with gamma energy greater than 0.5 Mev.

Activity of this order of magnitude would present formidable servicing problems and require remote handling techniques if the active products are not removed. Sanderson and Porter indicate that systems for removal of the kryp-

Table VII-3 SUMMARY OF IRRADIATIONS OF URANIUM GRAPHITE FUEL SPECIMENS*

Item	Reference and date reported	Where irradiated	Quant.	Type	U^{235} mg/cm^3	Enrich.	Form	Size	Temp. of irradiation	Fissions/cm^3	Notes†
1*					13.7	Th/U-11	UO_2	1½ in. sphere	2400°F max. 600°F min.	4×10^{19} 5×10^{18}	
2	Kierstead & Nagy; ANL-4006, ANL-4185; 1947	CP-3	23	Impreg.	Up to 25	Nat. & 30%	U_3O_8	0.4 cm dia. 0.4 cm L.	Low	Up to 2×10^{11}	a,b,c
3	Wohlberg; ANL-4427; 1949	HW	6	Impreg.	20 to 26	30%	U_3O_8	0.156 in. dia. 10 in. L.	100°C	Up to 2×10^{17}	b,d,f
4	Hunter; MonN-442, TID-66; 1947, 1949	X-10	1	Impreg.	300	93.6	UO_2	2 cm dia. 5 cm L.	775°C, 630°C	7×10^{18}	a
			1	Impreg.	300	93.6	UO_2	2 cm dia. 5 cm L.	630°C	3.5×10^{18}	a
			1	Admix.	300	93.6	UO_2	2 cm dia. 5 cm L.	630°C	5×10^{18}	a
5	Billington et al.; ORNL-576, 693 and 1095, TID-10042; 1949 to 1951	X-10							High and low		a
6*	Kernohan; ORNL-1722; 1954	X-10	12	Admix.	90	93.6	UO_2	¼ × 5/16 × 3 in.	30-88°C	2×10^{17}	a,b,c, e,f
7	Smith & Young; NAA-SR-72; 1951	LASL	1	Impreg.	2.1	Nat.	UO_2		Low	1×10^{12}	g
8	Cubicciotti; NAA-SR-194; 1952	LASL	Many	Impreg.	3.7	Nat.	UC_2	0.84 cm dia. 0.064-0.52 cm thick	Low	Up to 1×10^{15}	g
9	Doyle; NAA-SR-255; 1953	LASL	Many	Impreg.	1.4	Nat.	UC_2	5/8 in. dia-1/10 in. thick	Low	2×10^{14}	g
10	Young & Smith; NAA-SR-232; 1953	LASL	1	Impreg.	0.7	Nat.	UO_2		Low	1.5×10^{12}	g

11	McCarty & Steele; NAA-SR-223; 1953	HW	2	Impreg.	3.3	93.6	UO_2	1.3 in. dia × 4 in. L.	30°C	8.4×10^{17}	h
12	Hetrick et al.; NAA-SR-217; 1953	HW	2	Impreg.	24	93.6	UO_2	$1\frac{5}{16}$ in. dia × $4\frac{1}{4}$ in. L.	25-85°C surface 300-350°C center	7×10^{17}	a
13	Durand et al.; NAA-SR-836; 1954	MTR	1	Impreg.	35	93.6	UO_2	$\frac{1}{2}$ in. dia × 3 in. 3 in. L.	1300°C	1.1×10^{19}	a,b,d,i
			1	Impreg.	35	93.6	UO_2	$\frac{1}{2}$ in. dia × 3 in. 3 in. L.	1150°C	4.1×10^{18}	a,b,d,i
			1	Impreg.	35	93.6	UO_2	$\frac{1}{2}$ in. dia × 3 in. 3 in. L.	1000°C	4.7×10^{18}	a,b,d,i
14*	Moody et al.; ORNL-1778; 1955	X-10	Many	Lumped	18	93.6	UO_2	1 cm dia × 0.1 cm	Low	(3×10^{17} nvt)	g
15*	BMI-1181; 1957		13	Lumped		Nat.	UC_2		Low	(2.4×10^{15} nvt)	g
16	BMI; 1958	MTR	1	Admix.	40	93.6	UO_2	$\frac{3}{16} \times \frac{3}{16} \times$ 1 in.	1650°F	1×10^{19}	
			1	Admix.	40	93.6	UO_2	$\frac{3}{16} \times \frac{3}{16} \times$ 1 in.	1650°F	5×10^{18}	
17*	Cockcroft; Atomics & Nuclear Energy; 1958								600°C		
18	Krupp-BBC; 1958	NRX		Lumped		Nat.	UC_2	5 cm sphere	Low	(2×10^{20} nvt)	

*General comments. Item 1: Design conditions for 125 Mw(e) Pebble Bed Reactor (100 day; batch loaded). Item 6: Average UO_2 particle size 586, 334, 94, 20. Item 14: Many cladding types used, including SiC, TiC, and ZrC. Item 15: 0.03-in.-diameter UO_2 rod coextruded with graphite. Item 17, 0.25 per cent burn-up.

† Key for experimental information: a, thermal conductivity; b electrical resistivity; c, modulus of elasticity; d, dimensional change; e, weight change; f, postirradiation annealing; g, fission-product diffusion (by postirradiation heating); h, outgassing; i, crushing strength.

Table VII-4 IRRADIATION BEHAVIOR OF FUELED GRAPHITE SPECIMENS*

(Compiled from reference 5)

Specimen No.	Manufacturer†	Dia. decrease, %	Wt. change, %
11	BMI	0.26–0.33	−4.25
12	BMI	0.10–0.28	−0.71
E13	Sylcor	0.83–1.10	−4.48
E16	Sylcor	0.22–1.08	−2.88
86	Nat. Carbon	0.07–0.28	+0.03
87	Nat. Carbon	0.21–0.29	−0.08

* Specimens were 1½-in. fueled graphite spheres, packed in graphite flour in an irradiation capsule and operated at a surface temperature of ~1300°F to a burn-up of ~1½% of the 4.5 g of contained U^{235}.

†Each of the three companies manufactured two spheres: BMI by pressing powdered UO_2, carbon filler, and resin binder into 1-in.-diameter spheres, clad with a ¼-in. shell of resin bonded AGOT graphite and baked to set the binder; Sylcor by machining the spheres from AGOT graphite, fueled by soaking in uranyl nitrate solution, and roasted to convert the nitrate to UO_2; National Carbon by pressing a powdered mixture of graphite filler, pitch blender, and UO_2 and baking to set the binder.

Table VII-5 PER CENT OF ACTIVITY REMAINING IN URANIUM–GRAPHITE SAMPLES HEATED TWO TO THREE WEEKS AFTER IRRADIATION*

Element	2730°F (24 hr)	3100°F (24 hr)	3450°F (12 hr)
Cesium	4.5	1.4	
Strontium	1	0.07	0.1
Barium	6.5	1.5	0.07
Iodine	60	15	0.7
Tellurium	57	8	1
Pr (rare-earth group)	79	17	5
Yttrium	100	55	44
Cerium	100	76	30
Ruthenium	100	100	80
Molybdenum	100	100	100
Zirconium	100	100	100

* Data taken from reference 3.

Table VII-6 ACTIVITY EXPECTED IN PEBBLE BED REACTOR AT 350 MW(T)*

(Taken from reference 1)

Element	During operation	Activity in curies after shutdown: 1 hr	1 day	1 week
I	3.5×10^7	2.1×10^7	1.1×10^7	3.2×10^6
Xe	3.2×10^7	8.8×10^6	3.5×10^6	2.8×10^6
Kr	2.8×10^7	5.3×10^6	1.8×10^5	<350
Br	1.1×10^6	7×10^4	350	<350

* The activity curves shown in reference 1, Fig. 3.1-3, are apparently the activity per thermal megawatt and not the total activity. The data are for unclad or uncoated fuel elements assuming release of all elements volatile below 2500°F except Xe^{135}.

ton, iodine, and xenon are contemplated. However, should a coolant line break or if there is excessive leakage in the coolant system, a major radioactivity hazards problem will be created.

The estimated levels of activity are considerably lower if it is considered that most of the fuel will be operating at a temperature below 2000°F and that only part of the fission products will escape. For example, if it is assumed that the only fission products released from an unclad fueled graphite element are a result of fission recoil and diffusion through the graphite, then, for an element using 200 μ UO_2, about 4 per cent of the fission-product activity escapes from the UO_2 and diffuses through the graphite. The activity released, as shown by Cottrell,[6] at 2000°F would be approximately 5000 curies/Mw(t) or a total of 1.75×10^6 curies. This still results in an extremely high activity in the primary coolant loop.

A preliminary fission-product release test by BMI[5] of an irradiated fueled graphite specimen indicated that only 0.005 per cent of the Xe^{133} theoretically present had escaped. However, this result was reported to be subject to a large error due to uncertainties in the values of half life and other constants of Xe^{133} and because of the possibility that the loose graphite powder in which the specimen was packed may have adsorbed the gas at room temperature.

Because of the large activity to be expected from unclad elements of this type, the use of a protective coating to prevent the fission products from diffusing through the element would be advantageous. A further reason for applying some coating to the fuel elements comes from the necessity of preventing oxidation and other chemical reactions of the graphite by impurities in the coolant or by steam which may leak into the coolant of a reactor used with a steam generator. Several coatings are being actively developed by different manufacturers for the protection of graphite and to retain the fission products. Silicon carbide coatings have been widely used, although some difficulty in applying this coating is encountered because of the difference in the coefficients of expansion of SiC and graphite and consequent cracking of the coating. Silicon carbide will oxidize slowly in air at 3400°F and decomposes at 4000°F. Weight increases of about 0.15 per cent in 24 hr have been reported when SiC is exposed to 1000°C steam.

Minnesota Mining and Manufacturing Company has developed a coating known as 3M "Ceramic S" which is believed to be a siliconized silicon carbide. Tests performed by them indicate there is no weight loss in coated graphite specimens exposed 100 hr in 900°C steam or 150 hr in 900°C CO_2. These tests and the erosion tests conducted with air (see p. 43) are encouraging enough to warrant investigation of this type of coating in a reactor environment. Specimens with the coating are presently being irradiated. These tests will indicate whether the coating is stable and the effect of the higher stresses which may result because of the fission gas pressure build-up.

The Engineering Science Division of American Metal Products of Ann Arbor, Michigan, is developing diffusion-resistant coatings of graphite by diffusing carbides of hafnium, molybdenum, titanium, and zirconium into the surface of graphite. These coatings are reported to be good to 1000°C in air, although coated elements have been found to adhere to one another at 2600°C.

8. *Applicability.* If the uranium content of the graphite–uranium element is limited to the range which has been used to date, the elements may be limited to reactors burning highly enriched fuel, or to those employing thorium as the fertile material. The highest uranium content listed in Table VII-2 is about 30 wt.%. This corresponds to an average uranium density of about 0.67 g/cm^3 and to a graphite-to-uranium volume ratio of about 25. The latter is not less by a large factor than the average over-all graphite-to-uranium ratio which might be desired in a reactor core. Consequently, the graphite mixed with the uranium will constitute a sizeable fraction of the total core moderator, and its temperature will be an important factor in determining the neutron temperature in the fuel element. In such a case it may be expected that a prompt positive fuel-temperature coefficient of reactivity will develop, as plutonium builds up in the elements, because of the Doppler broadening of the plutonium fission resonance. To date there has been great reluctance to build reactors having such positive temperature coefficients.

Uranium-Magnesium Dispersions

Uranium metal in a dispersion with magnesium holds some promise as a fuel element. Fuel specimens containing about 8.5 wt.% magnesium, 50 vol.% uranium (approximately the same uranium content as in UO_2) have a thermal conductivity of approximately 35 Btu/(hr)(ft)(°F) at 1000°F, as compared to 1 Btu/(hr)(ft)(°F) for UO_2. These fuel specimens have been irradiated to burn-ups of 2.32 at.% uranium (~20,000 Mwd/ton) with good dimensional stability. Although magnesium has low corrosion resistance in water, the uranium-magnesium type fuel element has been successfully used by the USSR in their first atomic power station (a water-cooled graphite-moderated reactor) and is to be used in their superheater reactor (a boiling water and steam cooled, graphite-moderated reactor).[7,8]

The major advantages of the use of magnesium appear to be:

1. High radiation stability.
2. High thermal conductivity.
3. Low parasitic absorption cross section.
4. Good corrosion resistance to 490°F diphenyl.
5. Bonding by differential expansion of the magnesium with respect to the canning material.

The disadvantages are:

1. Lower volume of uranium in the fuel element.
2. Low corrosion resistance in water.
3. Relatively low melting point.
4. Diffusion into zirconium at elevated temperatures.
5. Possible metal-water or other autocatalytic reaction.

Some important properties of uranium, UO_2, and magnesium are shown in Table VII-7 for comparison purposes.

Table VII-7 SOME PHYSICAL PROPERTIES OF URANIUM, UO_2, AND MAGNESIUM

Material	Magnesium	Uranium (natural)	UO_2 (natural)
Density, g/cm^3	1.74	18.7	10.97 (theoret.)
Phase changes, °F	1200 (m.p.)	1229 (α–β) 1420 (β–γ) 2066 (m.p.)	5000 (m.p.)
Thermal conductivity, k Btu/(hr)(ft)(°F)	~85 (1000°F)	17 (1000°F)	~1 (>2000°F)
Thermal-neutron cross section			
σ_a (millibarns)	63	7680	7680
Σ_a (cm^{-1} × 10^3)	272	363	187

This type of element has been under study in AEC laboratories for several years but with only a minor development effort because of its low melting point. However, the USSR Atomic Power Station, APS-1,[9] using this type fuel element with water cooling, has been in operation since 1954. It is reported to have had only four fuel-element failures, which were handled with no difficulties. The tubular uranium-magnesium dispersion fuel element for the USSR superheater reactor is to operate at a maximum heat flux of 177,000 Btu/(sq ft)(hr) and a maximum uranium temperature of 1022°F.

Uranium-magnesium fuel specimens have been fabricated using uranium punchings, machine turnings, and metal shot with pure magnesium and magnesium-silicon alloy matrix materials. Densities as high as 65 vol.% uranium have been produced with uranium shot. Magnesium-silicon alloy as a matrix material was reported to be more desirable than pure magnesium since a bonding layer formed between the uranium and the magnesium-silicon alloy.[10] In addition, the corrosion resistance in boiling water of the uranium-magnesium-silicon dispersion was better than the dispersions in pure magnesium.

Specimens have operated stably at central temperatures of ~930°F (see Table VII-9), and test data indicate that the thermal conductivity of this fuel dispersion is over twice that of uranium metal.

Stability of this type element at very high burn-ups when clad with a strong restraining material appears to be very good. Test specimens have withstood burn-ups of 2.32 at.% uranium (~20,000 Mwd/ton) at a fuel operating temperature of about 730°F (see Table VII-9). In the Russian APS-I reactor tubular fuel elements of a uranium-molybdenum-magnesium dispersion have shown good performance since the reactor started operation in 1954. However, specimens clad in aluminum appear to be dimensionally unstable due to severe swelling. It would appear that, at the clad operating temperature of these specimens, the aluminum undergoes creep deformation under the stress due to the uranium distortion and the resultant plastic deformation of the magnesium.

Operation of this type element at high temperatures raises the question of compatibility with the clad material. Zircaloy-2-clad specimens have been operated at interface temperatures of ~592°F and to high burn-ups without failures. In these few specimens the diffusion of magnesium into the Zircaloy was not measured.

In a recent report[11] the diffusion of zirconium-magnesium couples was measured, and the following penetration times at various temperatures for magnesium into zirconium are listed in Table VII-8. Whether a barrier layer between the magnesium and the zirconium could be used is not known but might bear investigation.

The available irradiation data on uranium-magnesium dispersion fuel elements are listed in Table VII-9. Figure 5 is a plot of the weight per cent of magnesium used versus the volume

Table VII-8 TIME FOR MAGNESIUM TO PENETRATE 0.020 IN. INTO ZIRCONIUM*

Temp., °F	Time
392	92.45 years
572	2.63 years
752	9.38 weeks
932	113.3 hours

* Taken from reference 11, page 27.

Table VII-9 AVAILABLE DATA ON IRRADIATED URANIUM-MAGNESIUM DISPERSION FUEL ELEMENTS

SPECIMEN A*[10,12]			
No. specimens	2		
Composition	U–5 wt.% (Mg–0.4 wt.% Si)		
Vol. of U	64%		
Dimensions	~1 in. O.D. × 4 in. long		
Can material	Zircaloy-2		
Thickness	0.030 in.		
Burn-up, U atom %	0.2	0.35	1.0
Burn-up, Mwd/t	~2,000	~3,500	~9,900
Stability	Stable	Stable	Stable
Vol. % increase			0.5

* Specimen A. The above specimens were irradiated at a flux of $\sim 1.4 \times 10^{14}$ nv, operated at a heat generation rate of ~4.5 kw/cu in., with a central temperature of ~930°F and ΔT across the radius of ~338°F. At ~10,000 Mwd/t these elements are showing some swelling but are essentially stable.

Table VII-9 AVAILABLE DATA ON IRRADIATED URANIUM-MAGNESIUM DISPERSION FUEL ELEMENTS (Continued)

	SPECIMEN B*[10,13]				
No. specimens	5				
Composition	U–5 to 6 wt.% Mg				
Vol. of U	61 to 67%				
Dimensions	0.350 in. O.D. × 1.5 in. long				
Can material	Aluminum				
Thickness	0.136 in.				
	B-1	B-2	B-3	B-4	B-5
Burn-up, U atom %	0.1	0.1	0.3	0.3	0.3
Burn-up, Mwd/t	~1000	~1000	~1000	~1000	~1000
Stability	All specimens were unstable				
Vol. % increase	2.63	4.41	8.56	9.74	10.2

* Specimen B. Various sizes of uranium particles were reused in each of the above. Specimens B-1 and B-5 used 840–420 μ size; B-2 and B-3 a mixture of 60 wt.% 840–420 μ and 40 wt.% <44 μ; B-4 used 210–105 μ size. In all the specimens swelling was apparent even at the low burn-ups. Specimen distortion indicated uniform swelling which increased with exposure. This is what would be expected if the magnesium flowed plastically to redistribute uniformly the dimensional changes of the uranium and the canning material was not strong enough to restrain the pressure.

	SPECIMEN C*[10,12]			
No. specimens	12			
Composition	6 (U–8.5 wt.% Mg) 6 (U–8.5 wt.% Mg-Si)			
Vol. of U	~50%			
Dimensions	0.40 in. O.D. × 1.5 in. long			
Can material	Zircaloy-2			
Thickness	0.050 in.			
	C-1 to 4	C-5 to 8	C-9 to 10	C-11 to 12
Burn-up, U atom %	0.1	0.5	1.16	2.32
Burn-up, Mwd/t	1000	5000	10,000	20,000
Stability	All specimens were stable			

* Specimen C. These specimens were irradiated at a flux of $\sim 2 \times 10^{14}$ nv, operated at a heat generation rate of ~13 kw/cu in., with a core temperature of ~730°F. Specimens were stable even at 20,000 Mwd/t. Comparison with Specimen A would indicate that the extra restraint obtained with the thicker clad at the smaller diameter plus the lower operating temperature contributed to the better stability beyond 10,000 Mwd/t.

	SPECIMEN D*[7]	SPECIMEN E†[10]
No. specimens	USSR-APS-1 reactor elements (~512)	1
Composition	(U–9 wt.% Mo) 12 wt.% Mg	U–15 wt.% (Mg–0.4 wt.% Si)
Vol. of U	34%	35%
Dimensions	0.552 in. O.D. × 0.220 in. I.D. × ~67 in. long	1.4 in. O.D. × 4 in. long
Can material	S.S.	Aluminum
Thickness	0.0079 in. on O.D. 0.0158 in. on I.D.	0.050 in.
Burn-up, U atom %		0.08
Burn-up, Mwd/t	~18,000	~800
Stability	Stable	Unstable

* Specimen D. At the Second Geneva Conference the fuel element used in the first APS was described as a tubular uranium-molybdenum-magnesium dispersion element. Excellent stability was ascribed to this fuel element at high burn-up. It should be noted that a U–9 wt.% molybdenum alloy was used which in itself is a very stable alloy. In the superheat reactor which is expected to use an unalloyed uranium-magnesium element, the Soviets have only rated the fuel at 2300 Mwd/t. This may possibly be due to the high operating temperature (1022°F max.) of the fuel and the increase in the amount of fuel per element while maintaining the same thickness of stainless steel at the large diameter of fuel element.

† Specimen E. Here again the low strength aluminum clad resulted in swelling of the element with irradiation.

	SPECIMEN F*[12]	SPECIMEN G*[12]
No. specimens	2	2
Composition	U–22 wt.% Mg	U–35 wt.% Mg
Vol. of U	25%	15%
Dimensions	0.050 in. O.D. × 0.5 in. long	0.050 in. O.D. × 0.5 in. long
Can material	S.S.	S.S.
Thickness	0.010 in.	0.010 in.
Burn-up, U atom %	10.0	10.0
Burn-up, Md/t	>100,000	>100,000
Stability	Stable	Stable

* Specimens F and G. Specimens F and G used enriched uranium and operated at heat generation rates of 2.25 to 3.7 kw/cu in. at temperatures of ~572 to 752°F with no noticeable dimensional changes. With such small diameter elements, the 0.010 in. of stainless steel represents a strong clad material which, when combined with the much lower uranium content, results in a very stable element.

per cent of uranium metal obtained. All the irradiation test specimens listed in Table VII-9 are shown in Fig. 5.

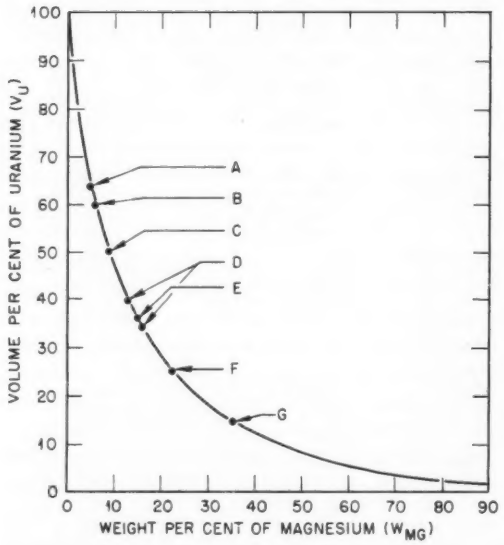

Figure 5—Stability of uranium-magnesium dispersion fuel elements. Specimen A, Zr-2 clad, stable to 1 U at.% (~10,000 Mwd/t). Specimen B, Al clad, unstable at >0.1 U at.% (~1,000 Mwd/t). Specimen C, Zr-2 clad, stable to 2.32 U at.% (~20,000 Mwd/t). Specimen D, S.S. clad, stable to ~2 U at.% (~18,000 Mwd/t). Specimen E, Al clad, unstable at 0.08 U at.% (~800 Mwd/t). Specimen F, S.S. clad, stable to 10.0 U at.% (>100,000 Mwd/t). Specimen G, S.S. clad, stable to 10.0 U at.% (>100,000 Mwd/t). For details, see Table VII-8.

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*Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, September 1958. (Individual papers available from the Office of Technical Services, Dept. of Commerce, Washington 25, D. C. Bound volumes of papers will be available from United Nations, Office of Public Information, New York.)

PROGRESS ON SPECIFIC REACTOR TYPES

BOILING REACTORS: THE PATHFINDER PROJECT

Work on the Pathfinder project was started by Allis-Chalmers Mfg. Co. in May 1957, under a contract with Northern States Power Co., as a result of the Third Round Power Demonstration Reactor Program.[1] The Central Utilities Atomic Power Associates (CUAPA), a nonprofit corporation of midwest utility companies, agreed to contribute $3,650,000 to the research and development effort, $1,200,000 of which was to be used for postconstruction research and development.[2,3] In response to a proposal by Northern States Power Co., the AEC in November 1957 also agreed to bear $6,000,000 of the research and development costs and to waive fuel use charges up to $1,000,000.[2] The Northern States contract was the first to be signed by the AEC under the Third Round Program.

The plant site selected is near the Big Sioux River, about six miles northeast of Sioux Falls, South Dakota. The construction schedule calls for the plant to go critical in April 1962 and into full power operation later in 1962.

As originally conceived,[4] the Pathfinder was to have been a controlled recirculation boiling reactor featuring a direct-cycle, forced-circulation, light-water moderator and coolant, and a separate conventionally fired superheater. The recirculation pumps were to be mounted inside the pressure vessel and to be driven at variable speed to control reactor power by changing the steam void content of the core. The reactor was to produce saturated steam at 600 psig and 489°F using UO_2 or UO_2-ThO_2 fuel jacketed in aluminum. The saturated steam was to be heated in a conventionally fired superheater and sent directly to a turbine generator. Early figures listed reactor thermal power as 133 Mw with steam temperature of 1000°F at the turbine throttle. During the course of the development, some modifications of these objectives have been made. The current design efforts are centered on the incorporation of nuclear superheat. The nominal thermal power of the reactor is 203 Mw, of which the superheater section is to produce 39 Mw. The reactor is to provide 600,000 pounds per hour of steam at 825°F and 540 psig to give a gross electric output of 66 Mw. Other design data for the plant are listed in Table VIII-1. The present estimated total cost of the project, including research and development and reactor construction, is about $33,100,000. Of this, the AEC will now furnish $8,500,000 for research and development and $1,800,000 for the waiver of fuel use charges.[5]

Since the work on the Pathfinder project includes the first extensive work which has been reported on integral superheat for boiling reactors, it will be reviewed in some detail. It is rather generally recognized that the outstanding advantage of the boiling reactor is its low capital cost per unit of steam rate. A major deficiency, shared by other water-cooled types, is the rather low temperature of the steam produced. It is also rather generally recognized that improvements are needed to reduce the fuel costs and the basic fuel utilization of water-cooled reactors. The use of integral nuclear superheat represents a possible means of removing one of these deficiencies. If this can be accomplished without causing substantial increases in reactor cost and fuel cost, it should represent a major improvement in the boiling reactor. A number of the problems which must be solved should be apparent from the following review of the Pathfinder work to date.[1,6-8]

Core Construction

The Pathfinder reactor core consists of a boiling section and a superheating section. The superheating section is located in the center and

Table VIII-1 PRELIMINARY DESIGN DATA FOR PATHFINDER PLANT WITH NUCLEAR SUPERHEATER

Plant	
Power, boiling region, kw	164,000
Power, superheating region, kw	39,000
Total core power, kw	203,000
Gross electrical capability, kw	66,000
Net electrical output, kw	62,000
Net efficiency, %	30.5
Steam outlet pressure, psig	540
Reactor operating pressure, psig	600
Temp., boiling region, °F	489
Outlet temp., superheating region, °F	825
Gross heat rate, Btu/kw-hr	10,370
Containment shell size, ft	50 × 115
Feedwater return temp., °F	300–360
Steam flow, lb/hr	600,000
Condenser pressure, in Hg abs	1.5
Reactor	
Vessel size	11 ft × 26 ft × 3 in.
Material, A 212, Grade B	$2\frac{3}{4}$ in.
Lining, 304L stainless	$\frac{1}{4}$ in.
Total core dimensions, ft	6 × 6
Dimensions of superheating region	6 ft × 30 in.
Fuel, boiling region (Al clad), enriched UO_2	~1.7%
Superheating region (S.S. clad), enriched UO_2	20% or more
Loading, boiling region, kg (U^{235})	100
Loading, superheating region, kg (U^{235})	24
Fuel, boiling:	
No. boiling assemblies	96
No. rods per assembly	81
Rod diameter, stepped	0.336 to 0.418 in.
Box size, square	5.218 in.
Fuel, superheating:	
No. superheating assemblies	429
No. tubes per assembly	2
Outer tube, O.D., and thickness	0.726 and 0.027 in.
Inner tube, O.D., and thickness	0.512 and 0.027 in.

is surrounded by the annular boiling section. Water is forced up through the boiling elements by pumps mounted in three recirculation loops connected to the pressure vessel. Steam produced in the boiling elements flows up through demister screens into the dome of the pressure vessel and enters a large downcomer connected directly to the superheating section of the core. After passing downward through the superheating section, the steam flows out of the pressure vessel through a section of insulated pipe which connects the lower end of the superheater section with a large nozzle in the bottom of the vessel.

Thermal isolation of the high-temperature superheater from the rest of the core is accomplished by mounting each superheater element in a double walled tube. The 57-mil annular gap between the double tube walls is filled with stagnant steam which acts as a thermal barrier. These tubes are arranged in a square pattern with a center-to-center opening of 1.236 in. and occupy a cylindrical section roughly 30 in. in diameter in the center of the core. The space between the tubes is filled with light water which serves as the moderator for the superheating region. The core is designed to avoid the formation of steam voids in this water, and the fuel-to-water ratio is selected to give a calculated minimum change in reactivity when the superheater is flooded with water.

Each superheater fuel assembly consists of two coaxial tubular elements, 36 mils thick, with an active length of 6 ft. The outer tube is 0.726 in. in outside diameter and the inner one is 0.512 in. in outside diameter. Each element contains enriched UO_2 dispersed in a stainless-steel matrix and is clad inside and outside with stainless steel (type 304L has been tentatively selected). Final choice of enrichment of the UO_2 has not been made but will apparently be either 20 or 93 per cent. The annular spacing of the tubes is maintained by spirally wound wire spacers 80 mils in diameter. The two fuel elements are welded at the top to form a single assembly which is inserted into the double walled tubes making up the superheater structure.

The boiling section of the core, which surrounds the superheater, is made up of rather conventional fuel boxes containing rod type elements filled with slightly enriched UO_2 pellets. A distinctive feature, however, is the use of 40-mil-thick aluminum alloy for the fuel jackets. A further novel feature incorporated in the boiling assemblies is the variation of fuel rod diameter along the length of the elements to help flatten the axial power distribution. Fuel rods will be 0.336 in. in outside diameter near the top and will be increased in steps to 0.418 in. in outside diameter at the bottom. For one configuration this approach gave a calculated axial peak-to-average power ratio in the boiling section of 1.21 to 1.

Many special aluminum alloys have been studied for the fuel jackets to find an alloy which will be usable at the reactor pressure of 600 psi. A material is needed having both high-temperature mechanical strength to withstand the pressures developed by fission gas release plus the necessary corrosion resistance in hot water. Composite tubes were tried in which the inner half of the tube wall was made of high strength 5083 alloy and the outer tube was made of corrosion-resistant X8001 alloy. Although excellent bonds were obtained in the composite

tubes, maintenance of required tolerances is difficult and expensive at present. Corrosion tests on X8001 and X2219 alloys have shown a rather high sensitivity to water velocity (see Table VIII-2). The limitations on velocity imposed by such characteristics may be a serious drawback in a forced-circulation reactor. The possibility that Zircaloy-2 rather than aluminum may have to be used is being considered.

Table VIII-2 EFFECT OF VELOCITY ON CORROSION OF ALUMINUM ALLOYS IN WATER AT 500°F

Flow rate, fps	X8001, mils/yr	X2219, mils/yr
0	1.3	0.9
7	6.2	6.0
18	8.8	8.0

Highly enriched superheating elements have been chosen for the reference design because a considerable background of proven technology exists for this type. The use of more economical, low-enrichment oxide elements is considered a future step which would require development. With enriched fuel in the center of the core, there is a large tendency for power to shift from the superheater section to the boiler section as burn-up progresses. This can be theoretically offset to some extent by the use of burnable poisons.

The central location of the superheater section, with its poor neutron economy, leads to rather low conversion ratios. The use of burnable poison would further reduce neutron economy. However, the central location can be advantageous in flattening the radial power distribution. In one case a radial peak-to-average ratio of 1.1 was calculated. The calculations show also that the flux pattern changes throughout core life, making it difficult to hold the power very flat.

The superheater is flooded with water during shutdown and at start-up, and the possibility always exists that accidental flooding might occur during operation. With the superheater in a region of high effectiveness in the core, it is essential that the fuel-to-water ratio be adjusted precisely to give a minimum change in reactivity in going from the steam-filled to the water-filled condition. For the Pathfinder core the total system change in reactivity with flooding of the superheater was estimated to be -0.7 per cent k_{eff} on the basis of flooding experiments conducted at Pennsylvania State University. Variations in the flooding coefficient with burn-up have not been discussed.

Control

The superheating reactor poses unique problems in control. Both the power level and the division of power between the boiling and superheating regions must be maintained at acceptable values. The present Pathfinder design has 16 cruciform control rods in the boiling region and 4 in the superheating zone. These rods have a blade span of 10 in. and an estimated total worth of about 17 per cent k_{eff}. This affords the minimum amount of control required for the core, and the use of burnable poisons is being considered to supplement the rods, especially in the superheater where large changes in reactivity occur with burn-up. Rod drives are of the rack and pinion type and are mounted on the top of the vessel.

Three methods have been considered for obtaining the variable coolant flow rate which is a feature of the design: (1) step control, (2) variable speed drive, and (3) throttling. None of these is completely without drawbacks. Step control, achieved by using two-speed motors on the pumps, has the obvious disadvantage of changing the flow abruptly in rather large increments. Variable speed electric drives seem to be expensive, and the amount of control available from throttling valves is limited by cavitation problems. Some difficulties have also been encountered in meeting the suction head requirements for the circulating pumps under all conditions, since in the Pathfinder design the net positive suction head (NPSH) obtained depends upon the cooling effect of adding feedwater to the saturated water from the reactor to lower the vapor pressure of the water being pumped. This was stated to be one of the chief reasons for abandoning the idea of mounting circulating pumps inside the pressure vessel. The internally mounted pumps required 50 ft NPSH whereas externally mounted pumps could be designed for operation with 20 ft NPSH.

Other Considerations

With an integral superheater the separation of steam and water must be accomplished within the pressure vessel. The Pathfinder design in-

cludes centrifugal separators to remove steam from the water being recirculated and demister screens to remove moisture contained in the steam. Except for the effects on pressure drop and space requirements in the vessel, these devices do not appear to present serious problems.

With high-temperature steam, it is possible that decomposition gases[9] may become a more serious problem for the turbine designer. The possibility of using a full flow catalytic recombiner in the main steam line is being studied in connection with the Pathfinder design.

Early in the project the possibility of avoiding the use of a full-sized containment shell for the Pathfinder was investigated. Although the problem is still under study, the reference design employs a cylindrical steel shell 50 ft in diameter and 115 ft high.

Preliminary studies of the dynamics of the Pathfinder plant indicate that the nuclear superheater should be quite stable, and with proper design of the superheater for flooding conditions no unusual hazards are anticipated. Emergency cooling of the superheater section is somewhat more difficult than for a conventional boiling reactor, since the superheater is normally steam cooled and in case of a scram the steam flow ceases. Calculations supported by experiment indicate that the superheater element can dissipate heat safely through the double walled tubes at 1 per cent of normal operating power.

Because of the relatively low temperature of saturated steam at 600 psig, it was originally hoped that a pressure sealing gasket might be used on the pressure vessel head. Recent tests of a silicone rubber seal resulted in complete rupture of the seal at 455 psi and ~475°F. The failed seal was described as bloated and tacky, bearing very little resemblance to the original gasket. Although alternate materials, such as Viton A, are being studied, other methods are now being sought for achieving a reliable, quick opening vessel closure.

References

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9. *Power Reactor Technology*, 2(2): (March 1959).

The design and development work which was done by Oak Ridge National Laboratory and by the Kaiser Engineers–American Car and Foundry group for the purpose of investigating the potentialities of the gas-cooled graphite-moderated reactor concept have been reviewed briefly in a previous issue of this Review.[1] In October 1958 an information meeting on gas-cooled power reactors was held at ORNL, and more recent work by Oak Ridge and the KE-ACF group was reviewed and several other gas-cooled reactor concepts were discussed.[2] A report issued by ORNL at about the same time contains a discussion of various advanced gas-cooled concepts which have been considered at Oak Ridge.[3] The following review covers briefly a new design of the KE-ACF plant and a number of other gas-cooled types which were discussed at the meeting.

The Kaiser-ACF Helium-cooled Reactor

The major change in the new Kaiser-ACF design[4–12] is the adoption of helium as the coolant. The prototype plant has a capacity of 30 Mw(e); the optimum plant has a capacity of 223 Mw(e). The design characteristics are summarized in Table IX-1.

The method of optimization for the helium-cooled plant was described in detail at the Information Meeting, and the calculated power generation costs for the optimized reactor were presented. The cost was 13.1 ± 2.5 mills/kw(e)-hr, and the detailed costs were about the same as those for the CO_2-cooled partially enriched uranium reactor reported in reference 1; the net cost difference is almost within the accuracy of the method used, stated to be ±0.25 mill/kw-hr.

The fuel elements for the helium reactors are almost identical to those proposed for the CO_2 reactor. The element consists of a cluster of seven 304 stainless-steel tubes about ¾ in. in outside diameter by 27 in. UO_2 pellets are contained in the stainless tubes and the seven-tube cluster is held in a graphite sleeve.

A critical item in the design of these types of fuel elements is the ability of the container tube to contain the fission-product gases. Table IX-2 gives calculated internal pressures in the KE-ACF fuel elements as a function of fuel exposure.[7] For coolant pressures of around 300 psia it can be seen that, for appreciable fission gas release, a fuel element must withstand buckling stresses at the start of its life and bursting stresses at the end of irradiation. Unfortunately the amount of gas released is not well known, and estimates vary from 1 to 100 per cent of that formed. ORNL has recently started a fuel capsule irradiation testing program, and some preliminary data are reported in reference 13. The experiments indicated

Table IX-1 SUMMARY OF KE-ACF HELIUM-COOLED POWER REACTOR DESIGN CHARACTERISTICS

	Prototype	Optimum
Rated thermal output, Mw	84.3	600
Rated net electric output, Mw	26.1	223
Net thermal efficiency, %	31	37.2
Specific power, Mw/metric ton		6.9
Enrichment, at.%		2.0
Uranium loading, metric tons		87
U^{235} inventory, kg		1740
Fuel material	UO_2	UO_2
Cladding material	304 stainless	Stainless
Av. exposure, Mwd/t	10,000	~10,000
Refueling	During operation	
Core dia. by height, ft	16 × 19	25 × 26.6
No. fuel channels	248	
Coolant pressure, psia	300	300
Coolant temp., °F:		
Inlet	450–500	430
Outlet	1050	1050
Coolant flow rate, lb/hr	~500,000	2,700,000
No. coolant loops	2	4
Pressure vessel I.D. height, ft	20 × 40	
Steam cycle	Dual pressure reheat	Dual pressure reheat or no reheat

Table IX-2 CALCULATED FUEL-ELEMENT INTERNAL PRESSURE VS. FUEL EXPOSURE[7]

(Heat generation; 24,100 Btu/(hr)(ft); clad temp., 1300°F)

Fuel-element internal pressure, psia	Exposure, Mwd/t
Fission gas release fraction	0.10
75	0
125	5,000
160	10,000
200	15,000
Fission gas release fraction	0.50
300	5,000
510	10,000
700	15,000
Fission gas release fraction	1.0
460	5,000
850	10,000

that, at an internal temperature of 1770°F, exposures of 6000 and 12,000 Mwd/ton resulted in a fractional release of 1.7 per cent and about 2 per cent, respectively. Although these fractional releases are low, the temperature is also; the clad temperature of 1300°F given in Table IX-2 is associated with a UO_2 outer surface temperature of about 2300°F and a centerline temperature of over 3000°F.

The Pebble Bed Reactor

Two recent reports[14,15] describe studies of the pebble bed reactor concept. The basic principle of the concept is that fuel and some or all of the moderator be intimately mixed and formed into elements of simple shape (e.g., spheres) which may then be simply dumped into a suitable vessel to form the reactor core. Cooling of the elements is by gas flow in whatever spaces are left vacant between elements. To a very large extent the feasibility and attractiveness of such a reactor are determined by the performance of the fuel-moderator elements. A discussion of elements of types considered for the pebble bed reactor is given in Sec. VII of this Review.

The reactor described in references 14 and 15 consists of a pressure vessel containing graphite structural parts which form seven central vertical channels for the fuel elements and an annular ring channel for the blanket elements. The blanket and fuel elements are unclad graphite spheres 1½ in. in diameter containing fissionable and/or fertile material. Helium is pumped through the reactor as the heat exchange medium.

By the use of graphite and helium, quite high temperatures are expected to be possible. Other advantages which are expected to result from the pebble bed concept are low fuel fabrication and handling costs, the use of relatively simple equipment, and a possibly greater facility than clad elements.

The disadvantage of the system in its present form is that fission products are present in the primary loop because of the unclad fuel elements. This high level of radioactivity (Sec. VII) may be expected to give serious maintenance problems and perhaps operational problems. Some of the activity may be removed from the loop by compound formation, adsorption, or refrigeration. In the design of the heat exchanger an attempt is made to minimize the amount of maintenance which may be required. Excess tubes are provided in the exchanger; should a steam tube burst, the ruptured tube can be plugged from the outside.

Interesting features of the system are the methods used to minimize expansion problems and to protect the pressure vessels from the hot gas. The reactor, heat exchangers, and blowers are mounted in the same plane in order to reduce the expansion difficulties. The gas flows between the reactor and the heat exchangers in coaxial double-wall pipe, with the hot gas flowing in the center and the cooler gas flowing in the outer annular space. In both the reactor and the heat exchangers the cooler gas is used to sweep the walls, thus maintaining their temperatures near the lowest temperatures of the system.

The uses of this system as a U^{235} burner, U^{233}-thorium converter, U^{233}-thorium breeder, or U^{235}-plutonium converter are being considered. With U^{233} as the fissionable material and thorium as the fertile material, a conversion ratio of 0.846 or higher has been estimated. On this basis, power costs were estimated to be between 7.61 and 8.76 mills/kw-hr.

Some pertinent plant data appear in Table IX-3.

The Gas-cooled D_2O-moderated Reactor

The gas-cooled D_2O-moderated reactor[16,17] project is a research and development program,

Table IX-3 PEBBLE BED REACTOR PLANT DATA, 125 MW(E)

General	
Reactor output, Mw	337.15
Heat from blowers, Mw	7.67
Heat to steam generators, Mw	344.82
Capability, Mw-gross/net	139.03/126.75
Auxiliary power, Mw	12.28
Heat rate, Btu/kw-hr gross/net	8464/9077
Containment Vessel	
Size	75 ft dia. × 94 ft high
Wall thickness, in.	1
Design pressure, psig	35
Shielding	Concrete
Thickness, ft	1 and 2½
Reactor Vessel	
Size	13 ft 8 in. × 21 ft
Wall thickness, in.	5.5
End thickness, in.	2.75
Thermal shield thickness, in.	2
Reactor Core	
Equivalent dia., ft	9
Height, ft	8.1
Voidage, %	29
Moderator and structure	Graphite
Fixed moderator volume, cu ft	130
Ball bed volume, cu ft	383
Reactor Blanket	
Equivalent thickness, ft	1.33
Height, ft	8.1
Voidage, %	39
Ball bed volume, cu ft	496

Thermal and Fluid Dynamic Characteristics	
Design power level, Mw	350
Design pressure level, psia	1000
Design helium flow, lb/hr	1,360,000
Volumetric heat generation, Mw/cu ft ball bed	0.915
Temp., °F:	
Reactor inlet	550
Reactor outlet	1250
Core outlet, av.	1540
Blanket outlet, av.	561
Fuel-element surface, max.	2170
Fuel-element center, max.	2440
Operating power level, Mw	337
Core contribution,* Mw	291.5
Blanket contribution,* Mw	45.5
Operating pressure level, psia	965
Operating helium flow, lb/hr	1,343,000
Core pressure drop, psi	15.5
Helium Blowers (3)	
Capacity	20,000 cfm
Main drive	3500 hp, 3600 rpm
Emergency drive	30 hp, 450 rpm
Total pumping power	8536 kw

Steam Generators (3)	
Size	8 ft 6 in. dia., 36 ft high
Steam pressure and temp.	1450 psig, 1000°F
Steam flow per generator	352,000 lb/hr
Feedwater temp.	400°F
No. feedwater heaters	4
Helium temp.	1207°F in; 500°F out
Control Rods	
No.	12
O.D., in.	4
I.D., in.	3½
Material	Haynes-25
Length, in.	99
Weight, lb	83
Drive	Electric motor
Reactivity worth, initial	17.3%

Fuel Elements

	Core	Blanket
Form	Spherical	Spherical
Dia.	1.5 in.	1.5 in.
No.	228,600	296,250
Composition by weight, g:		
Graphite	48.49	45.28
ThO_2	4.94	45.28
UO_2	0.45	
Total	53.88	90.56

Nuclear	
Core C/Th/U atom ratio	3745/11/1
Blanket C/Th atom ratio	22/1
Critical mass (1200°F, eq. Xe), kg U^{233}	60.2
U^{233} loading, kg	90.2
Th^{232} loading in core, kg	992.2
Th^{232} loading in blanket, kg	11,780
Fuel burn-up rate, kg/Mw year	0.48
Initial capture to fission ratio	0.257
Fraction of epithermal fissions	0.372
Initial breeding ratio, core	0.477
Initial breeding ratio, blanket	0.353
Total	0.830
Av. breeding ratio over core lifetime	0.863
Av. power density, watts/cm^3	23.7
Atom burn-up (% total atoms):	
Core (after 100 days)	0.021
Blanket (after 1000 days)	0.006
Temp. coefficient	-2.7×10^{-5}/°F
Initial multiplication factor:	
k_{eff} (1200°F, eq. Xe)†	1.100
k_{eff} (77°F, no Xe)	1.204
Av. thermal flux at start-up, neutrons/cm^2-sec	1.3×10^{14}
Av. thermal flux, after 100 days, neutrons/cm^2-sec	2.2×10^{14}

* At equilibrium conditions, after 10 core lifetimes.
† Equivalent bare homogenized core model.

sponsored by the East Central Nuclear Group and the Atomic Energy Commission, which is to lead to the construction of a 50 Mw(e) plant by the Florida West Coast Nuclear Group. The plant design is being made by the General Nuclear Engineering Corporation and the American Electric Power Corporation. Initial operation is scheduled for 1963.

Table IX-4 CORE AND FUEL DATA, GAS-COOLED D_2O-MODERATED REACTOR[16]

Fuel material	UO_2 enriched to 1.84%
Clad material and thickness	Stainless steel, 0.015 in.
Fuel shape	Solid UO_2 or hollow (cooling on outside only)
Dia. fuel	0.20 in. I.D., and 0.54 in. O.D.
Rods per bundle	19
Length of bundle, in.	18
No. bundles per tube	8
UO_2 load, short tons	17.4
Insulating material	Ceramic
Lattice pitch, in.	8
Pressure tube	Zr-2 (0.125 in. min.)
Size of pressure tube, I.D., in.	4.75
No. pressure tubes	178
Core active length, ft	12
Core dia., ft	10
No. control blades	24
Width of control blade, in.	10
Tons of D_2O in reactor tank	44.5
Pressure of D_2O tank, psig	15
Av. temp., °F	217

Table IX-5 CORE PERFORMANCE, GAS-COOLED D_2O-MODERATED REACTOR[16]

Physics of 8-in. lattice:	
Enrichment (at.%)	1.84
k_∞ (hot, Xe, and Sm)	1.205
η	1.705
ϵ	1.019
p	0.839
f	0.831
k_{eff} (hot, Xe, and Sm)	1.072
Available for burn-up and operation	0.072
Initial conversion ratio	0.54
k_{eff} (cold, clean, and completely loaded with 1.84% enrichment)	1.156
Av. heat flux, Btu/(hr)(sq ft)	95,300
Max. heat flux, Btu/(hr)(sq ft)	262,000
Pressure of CO_2, psig	~500
Gas velocity in max. power tube, ft/sec	112.5
Pressure drop through core, psi	13.2
Max. surface temp., °F	1500*
Max. center temp., °F	3800
Outlet gas temp., °F	1050
Inlet gas temp., °F	500
Mw/t of UO_2	10.0
Mw/t of D_2O in tank	3.8

* 100°F allowance for coolant hot spots.

The reactor is of the pressure tube type, using low-temperature D_2O as moderator, and is to produce gas temperatures of about 1000°F. It is to be a prototype for a large-scale plant (200 Mw(e) range) of high neutron economy. The prototype will initially be fueled with uranium oxide jacketed in stainless steel, but all stationary core components other than the fuel jackets are to be of low cross-section materials compatible with the goal of high neutron economy for the full-scale reactor. Physics calculations indicate that a large reactor having a core construction like that of the prototype, but employing beryllium-jacketed oxide fuel, would operate on natural uranium to give a fuel lifetime of the order 4000 Mwd/ton.

The prototype reactor consists of a low-pressure tank of D_2O traversed by 178 zirconium pressure tubes which contain the fuel elements and the coolant gas, CO_2. The pressure tubes are thermally insulated on the inside and are cooled outside by the D_2O moderator. The tubes terminate at each end in steel box type headers which are joined together by bellows to allow for thermal expansion. The fuel elements consist of bundles of 19 fuel rods approximately 18 in. long. The fuel rods are made up of cylindrical UO_2 slugs jacketed in 0.015-in. stainless steel. The characteristics of the prototype core and fuel are given in Table IX-4, and the performance characteristics of the core are given in Table IX-5.

The outstanding advantage of the gas-cooled D_2O-moderated reactor is its potentiality for the simultaneous attainment of high neutron economy and high coolant temperature. It is probably the only gas-cooled type which has the possibility of operating with uranium oxide fuel of natural enrichment, and although there is some argument as to whether that type of operation is most economic, there can be little question of the advantage of the high neutron economy, whether operation be with natural fuel, enriched fuel, or recycled fuel. The reactor type requires the solution of several difficult developmental problems. The most important of these appear to be the insulation of the pressure tube, the joining of the zirconium pressure tubes to the steel headers, and the development of the fuel element.

The Marine Gas-cooled Reactor Program

The Marine Gas-cooled Reactor (MGCR) program was initiated on Feb. 17, 1958, as a contract between General Dynamics Corporation and the U. S. Atomic Energy Commission's Maritime Reactors Branch. The specific objective of the MGCR program is to develop, design, build, and test a prototype closed-cycle gas-cooled reactor and gas-turbine power plant. The broad objective is to expand and advance the technology of gas-cooled systems and components, particularly in the temperature range above 1300°F.

Reference 18 describes the more conclusive results obtained to date by General Atomics on the reactor materials program and the principal considerations that led to their selection of a helium-cooled graphite-moderated reactor as having the greatest potential relative to the MGCR as well as other applications of high-temperature gas-cooled reactors.

The thermal output of the prototype reactor will be about 49.7 Mw and will supply 20,000 hp to the propeller shaft and 750 hp for auxiliary ships load. Helium gas at 791°F and 971 psi will enter the reactor through the annulus of coaxial pipes, pass between the moderator and the pressure vessel, pass downward across the fuel, and exit at 1300°F through the inner coaxial pipe.

The principal reason for use of helium as a coolant in the General Atomics design is the likelihood that it can be used in conjunction with unclad graphite moderator. The helium will probably have its impurities removed to minimize graphite reactions.

Reference 18 describes four moderator-coolant systems that were investigated. They are helium-graphite, CO_2-graphite, CO_2-water, and CO_2-zirconium hydride. This investigation included tests on graphite-metal and CO_2-metal compatibilities with the metal temperature at 1500°F.

The results of the graphite-metal tests for various cladding materials were:

1. High-strength nickel-base alloys and the austenitic stainless steels carburized rapidly at 1500°F. At 1300°F, only two of the alloys tested (types 347 and 310) carburized.

2. Ferritic stainless steels (except type 405) and the iron-chromium-aluminum alloys resisted carburization for 716 hr at 1500°F. These materials are subject to grain growth, however, and have virtually no strength at this temperature.

3. One mil of chromium plate or diffused aluminum was effective as a carburization barrier.

These carburization test results indicate the cladding difficulties that would be encountered for a CO_2-cooled graphite system or for fuel-element cladding when the fuel contains graphite. The chromium and diffused aluminum coatings have yet to be evaluated under thermal cycling and stress conditions. The ferritic alloy-graphite reaction must be checked under irradiation conditions.

The CO_2-metal compatibility test was made with static CO_2 at 2000 psi. The results were based on 1600-hr tests.

Some of the results of the 1600-hr CO_2-metal test are:

1. All materials tested had low over-all corrosion rates.

2. Austenitic steels are subject to localized intergranular attack and pitting.

3. Type 316 and the Inconels should be considered as strong possibilities for high strength fuel cladding.

Further data are required, however, to establish the feasibility of any of these materials at high temperatures in CO_2. General Atomics has initiated dynamic corrosion testing to evaluate further the more promising alloys.

The water-moderated reactor requires physical barriers between the water and the gas, and this barrier must be insulated from the gas stream. However, it is considered desirable for the fuel element to be able to lose enough heat to the moderator to prevent fuel failure when coolant flow is lost; in such a case the normal heat loss to the moderator through the insulating shrouds is allowed to be as high as 12 per cent of the total thermal power.

The hydride-moderated reactors considered were of two types: one operating with a thermally "hot" moderator cooled with reactor inlet gas, and one operating with a "cold" moderator cooled by means of an auxiliary gas circuit. The first arrangement was recommended on the basis of simplicity, and the maximum hydride temperature was limited to 1000°F to keep the hydrogen partial pressure low. This was accomplished by insulating the

process tubes; in addition, cladding of the hydride was deemed necessary to prevent escape of hydrogen. The high power density in the hydride reactor causes emergency cooling problems, particularly since the process tubes are insulated. Radiant heat transfer was found to be insufficient to cool the fuel rods in the case of loss of coolant.

Both heterogeneous and solid homogeneous fuel elements are considered by General Atomics for the reference design. A typical heterogeneous fuel assembly consists of a bundle of 19 stainless-steel rods about 0.25 in. in diameter and 12 in. long. Six such bundles are connected to form an assembly about 6 ft in length and are positioned vertically in a cylindrical process tube in the graphite. There would be about 350 such tubes. The fuel is visualized as UO_2 but dispersed in a diluent which may be BeO, Al_2O_3, MgO, CeO_2, ZrO_2, or graphite. The diluent is needed to achieve sufficient heat-transfer surface without exceeding the ratio of U^{238} to moderator which is desired for resonance escape reasons. A problem with this fuel element is the rapid temperature rise in case of loss of coolant flow. The heterogeneous fuel rod of clad UO_2 generally has little heat capacity, and if coolant flow is stopped, it must depend principally on radiation to prevent a rapid temperature rise. The solid homogeneous rod with a large dilution of the fuel by graphite, in comparison, has a high heat capacity which retards the temperature rise. The homogeneous design employs steel-clad slugs, about 2 in. in diameter, placed in a close-pitch lattice. An attractive alternative considered consists of a calandria type element in which a somewhat larger slug is cooled by gas passing through several tubes running the length of the element. Furthermore, the large dilution of fuel with graphite should exhibit good burn-up properties since, for 30 per cent burn-up of the U^{235}, only 0.02 per cent of the total atoms in the fuel body are fissioned. Graphitization of the clad material at 1500°F seems to be the principal problem for the graphite-uranium element.

Comparison of the four different gas-moderator systems indicates that the principal element in determining the fuel cycle costs was the ability to operate at high burn-up. The comparison indicated that the helium-graphite type has slightly lower fuel costs than the CO_2-graphite type reactor.

Much of the shielding is incorporated as secondary shielding, integral with the engine room containment, in the General Atomics design; the secondary shield contribution to the total weight was comparable to, or greater than, that of the primary shield for the four gas–moderator combinations considered. It represents 41.5 per cent of the 976 ton total.

The gas-cooled reactor which was selected by American-Standard in their design study of nuclear ships[19] is similar to the General Atomics design, in that (1) it is helium-cooled and graphite-moderated, and (2) gas enters the outer annulus of coaxial pipes at 783°F and 970 psi, passes over fuel, and exits at 1300°F through the inner pipe. The maximum clad temperature, however, is 1600°F. The graphite moderator in the American-Standard design is clad with stainless steel, and the dominant reason for using helium is given as the induced activity in the CO_2 as compared to the negligible induced activity of helium. The American-Standard design differs, however, in the fuel-element geometry, employing a number of small-diameter UO_2 rods clad in steel and assembled in an annular ring. The ring is placed in a block of graphite, and the moderator is also present as a central plug.

Table IX-6 compares the principal characteristics of the four designs considered by General Atomics.

Although detailed cost data were not given, it was stated that "no significant differences" were found between plants. Since no significant differences are apparent in immediate economic performance, taking into account the factors of plant weight, plant cost, and fuel cycle costs, the choice of moderator and coolant then depends upon the system showing the greatest likelihood of success in achieving the technical performance for the MGCR and greatest potential for subsequent development toward systems of even higher performance. Graphite appears to be the most promising moderator from the point of view of safety and ability to meet the design conditions, but these features of graphite can only be relied upon through the choice of helium.

The selection of helium and graphite avoids the uncertainties of clad operation in CO_2 and the graphitization of clad on the moderator. It also avoids some of the problems of the loss of coolant accident that characterize the water- and hydride-moderated reactors.

Table IX-6 COMPARISON OF POSSIBLE MARINE GAS-COOLED REACTORS[18]

(Shaft output, 20,000 hp)

	Graphite Helium	Graphite CO_2	H_2O CO_2	ZrH CO_2
Net plant efficiency, %	30.0	32.5	31.4	33.1
Reactor heat output:				
Heat input to coolant, Mw	49.7	45.9	44.7	44.7
Heat loss to moderator, Mw			2.8	
Max. coolant pressure, psia	1000	1000	1000	1000
Exit temp. of coolant from reactor, °F	1300	1300	1300	1300
Dia. and height of reactor core and reflector, ft	8.37	8.37	5.08	5.00
Reactor vessel dia., ft (O.D.)	10.4	10.4	7.43	7.28
Reactor vessel height, ft	21.1	21.1	16.7	15.0
Reactor vessel wall thickness, in.	5.75	5.75	7.58	7.44
Reactor weight, tons	173	173	80	99
Primary shield, tons	572	566	410.5	353
Secondary shield, tons	404	438	470	470
Moderator cooling system, tons			7.5	
Turbomachinery, tons	15.8	17.1	14.3	14.3
Regenerator, tons	37.1	71.5	60.7	60.7
Intercooler and precooler, tons	27.0	38.2	26.9	26.9
Valves and piping, tons	26.8	35.7	30.4	30.4
Coolant storage system, tons	24.7	35.2	29.0	29.0
Reduction gear, tons	134.0	111.5	175.0	175.0
Fluid load, tons	235.3	244.5	241.6	241.6
Auxiliary systems and equipment, tons	367.8	377.8	369.1	367.0
Total machinery weight, tons	2017.5	2108.5	1915.1	1869.1
Specific weight, lb/SHP	202	211	192	187

The Gas-cooled Reactor Experiment

The Gas-cooled Reactor Experiment (GCRE) is part of the program of the Army Reactors Branch of the Atomic Energy Commission. Aerojet-General Nucleonics has prime responsibility for the design, fabrication, and operation of the GCRE-I, now being built at the National Reactor Testing Station. It is to develop engineering data and provide experience for mobile power plants for military use and small stationary power plants for civilian use.[20] Aerojet-General also has begun the preliminary design of a mobile plant which will be able to operate under extreme environmental conditions and which is transportable intact by trailer or cargo aircraft. Construction is scheduled to start[20] in January 1960. A further item in the Army gas-cooled program is GCRE-II, a backup and advanced design for GCRE-I. It also is being carried out by Aerojet-General.

The objective of the Army gas-cooled program is similar to that of the MGCR program in that it is directed toward a closed-cycle gas-turbine plant. It differs, however, in that the emphasis is on portability, operation under extreme environmental conditions, and independence of any special supporting facilities of either natural (e.g., cooling-water sources) or man-made origin. Presumably, also, there is less emphasis on costs.

The GCRE-I core[21,22] consists of coaxial tubular fuel elements which are composed of stainless steel and UO_2 "meat" clad with stainless steel. The coaxial fuel tubes are contained in insulated aluminum pressure tubes through which nitrogen coolant gas flows. The reactor assembly (73 pressure tubes and 2 plenum chambers) is immersed in a pool of water. The water acts as the moderator, the primary radiation shield, and the external core coolant. A lead fast-neutron reflector and gamma shield surrounds the core assembly. In normal operation at 2.2 Mw the nitrogen coolant is heated from 800 to 1200°F as it passes through the core. Under these conditions the maximum fuel-element temperature (including "hot spot" factors) is 1650°F.

The fuel elements will be formed from plates of UO_2 dispersed in type 316 stainless steel and clad with type 318 stainless steel. The plate is 0.045 in. thick with a 0.033-in. core of 30 wt.% UO_2-stainless steel and a 0.006-in. cladding. These plates are rolled to form 120-deg sectors of the cylindrical fuel tubes; each fuel tube is made up of three of these rolled sections.

The fuel element will comprise four coaxial fuel tubes contained in an aluminum pressure tube. The spacing of the fuel tubes is varied to provide larger coolant passages where the heat generation is the greatest. A layer of insulating material will provide a thermal barrier between the coolant gas and the pressure tube.

Metal liners will be provided on each side of the fuel-element insulation. The active portion of the fuel elements will be 1.25 in. in diameter and 28 in. long. The pressure tube is 1.875 in. in outside diameter with 0.058-in. wall.

The nitrogen gas coolant called for in the GCRE-I design is a specification peculiar to this program which results from the specific objectives of the program. Other gas-cooled reactor projects have avoided nitrogen for several reasons, one of the most important being that materials contacted would become nitrided, therefore brittle, and possibly fail. Experiments sponsored by this program at Battelle Memorial Institute indicate that nitriding can be inhibited by small amounts of additives in the coolant gas stream.

The GCRE design calls for a maximum temperature of 1650°F for the stainless-steel fuel clad. This is considerably higher than most power reactor projects have set as their maximum design limit for stainless-steel fuel cladding. Some of the results of the GCRE test program appear to indicate that AISI type 318 stainless-steel cladding will operate at 1650°F in the GCRE nitrogen atmosphere for the required 3000- to 10,000-hr operating time. This high design temperature may, however, be overly optimistic. Tests conducted on fuel elements developed by the Martin Company to evaluate their possible application to GCRE gave the following results. After periods of 1500°F exposure, the AISI type 316 stainless-steel outer cladding showed severe intergranular type corrosion. Penetrations 0.0035 in. deep were common, even though the total exposure time was not of extended duration. Additional tests to determine why this occurred will be performed to verify these results if a serious interest in these elements continues.

The critical experiments for the GCRE program are being done by BMI.[23,24]

The Nuclear Gas Engine

References 25 and 26 describe a new concept in nuclear power plants, the nuclear gas engine, proposed by A. P. Fraas, of ORNL. The device consists of a reciprocating engine operating on a gas cycle and connected to a gas-cooled reactor. During the compression stroke, cool gas is forced from the cylinder into the reactor through a valve. At, or near, the top of the

Table IX-7 SPECIFICATIONS FOR THE PROPOSED NUCLEAR GAS ENGINE POWER PLANT (NGE-1)

Normal power for propulsion	20,000 SHP
Max. power for propulsion	22,000 SHP
Power to electrical system	1500 kw
Reactor power (max.)	60 Mw(t)
Working fluid	Dry N_2
Over-all thermal efficiency	30%
Reactor data:	
Fuel element	UO_2 in S.S. capsules
Moderator	Graphite, grade TSF
Lattice pitch	8.0 in.
Core dia.	8.5 ft
Core length	10.0 ft
Side reflector thickness	1.5 ft
End reflector thickness	2.0 ft
Fuel channel dia.	2.9 in.
No. fuel elements required	4655 capsules, 665 assemblies
No. fuel channels	133
Pressure vessel I.D.	12 ft 8 in.
Pressure vessel O.D.	14 ft
Pressure vessel height	23 ft
Shield O.D.	26 ft
Shield height	32 ft
Engine data:	
No. main propulsion engines	4
Bore	15.5 in.
Stroke	22 in.
No. cylinders	12
rpm	335
Compression ratio	4.8
BHP per engine	7,500
BMEP	130 psi
Piston speed	1230 ft/min
Piston displacement (per engine)	49,800 cu in. (28.9 cu ft)
Gas system data:	
Total gas system volume	1800 cu ft
High pressure system	1200 cu ft
Low pressure system	600 cu ft
Volume of free gas at STP	50,000 cu ft
Gas flow through reactor at full power	125.5 cu ft/sec 465 lb/sec
Gas velocity through reactor at full power	90 ft/sec
Gas velocity through ducts (mean)	75 ft/sec
Gas temp. at reactor inlet	600°F
Gas temp. at reactor outlet	1100°F
Gas temp. at cooler inlet	470°F
Gas temp. at cooler outlet	90°F
Pressure at cylinder discharge port	1500 psi
Pressure at cylinder return port	1456 psi
Pressure at cooler inlet	157 psi
Pressure at cylinder intake port	165 psi
Gas density at cylinder discharge port	3.7 lb/cu ft
Gas density at cooler outlet	0.723 lb/cu ft

stroke the exhaust valve closes and an intake valve opens, admitting hot gas to the cylinder. At the bottom of the stroke several ports are uncovered and a blower forces cool gas into the cylinder, displacing the expanded gas through the ports, through a cooler and to the blower. The system is proposed as a possibility for ship propulsion and similar applications.

The major reason for considering this cycle was the desire to obtain a high thermal efficiency in a closed gas cycle at lower reactor coolant temperatures than those which characterize the gas-turbine cycle. Preliminary study indicated that a cycle efficiency of about 30 per cent could be obtained with a nominal system pressure of 100 atm and a reactor coolant exit temperature under 1200°F, employing nitrogen as a coolant. Other pertinent information is summarized in Table IX-7.

The major problems in development of the cycle were mentioned as follows:

1. Minimizing breathing losses in the engine.
2. Proper design of engine cylinder head and valve gallery.
3. Minimizing leaks.
4. Carry-over of lubricating oil from the engine.
5. Afterheat removal.
6. Operation, control, and start-up.

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